NEXTera® ENERGY SEABROOK

March 31, 2016

10 CFR 50.90

SBK-L-16029

Docket No. 50-443

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Seabrook Station

License Amendment Request 16-01
Request to Extend Containment Leakage Test Frequency

In accordance with 10 CFR 50.90, NextEra Energy Seabrook, LLC (NextEra) is submitting License Amendment Request (LAR) 16-01 to revise the Seabrook Station Technical Specifications (TS). The proposed change would revise TS 6.15, Containment Leakage Rate Testing Program, to require a program that is in accordance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

Attachment 1 to this letter provides NextEra's evaluation of the proposed change. Attachment 2 provides a markup of the current TS showing the proposed change, and Attachment 3 contains the revised (clean) TS page. Attachment 4 provides the plant-specific risk impact assessment, and Appendix A to the attachment provides documentation of the technical adequacy of the Seabrook Station probabilistic risk assessment.

As discussed in the evaluation, the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the change. This change has been reviewed by the Seabrook Station Onsite Review Group.

In accordance with 10 CFR 50.91, NextEra is notifying the State of New Hampshire of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

No new commitments are made as a result of this change.

NextEra requests NRC review and approval of LAR 16-01 by March 31, 2017, and implementation within 30 days.

A001
NRR

Should you have any questions regarding this letter, please contact Mr. Michael Ossing, Licensing Manager, at (603) 773-7512.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 31, 2016.

Sincerely,

Dean Curtland
Site Vice President
NextEra Energy Seabrook, LLC

Enclosure: Evaluation of the Proposed Change

cc: NRC Region I Administrator
NRC Project Manager
NRC Senior Resident Inspector

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ATTACHMENT 1

NEXTERA ENERGY SEABROOK STATION, LLC
SEABROOK STATION

LICENSE AMENDMENT REQUEST 16-01
REQUEST TO EXTEND CONTAINMENT LEAKAGE TEST FREQUENCY

EVALUATION OF PROPOSED CHANGES

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1.0 SUMMARY DESCRIPTION

NextEra Energy Seabrook, LLC (NextEra) requests a license amendment to revise the Seabrook Station Technical Specification (TS). The proposed change would revise TS 6.15, Containment Leakage Rate Testing Program, to require a program that is in accordance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 6.1). The NRC determined that NEI 94-01 describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, as modified by the conditions and limitations in its Safety Evaluation (Reference 6.2). This proposed change will allow extension of the Type A test interval up to one test in 15 years and extension of the Type C test interval up to 75 months, based on acceptable performance history as defined in NEI 94-01, Revision 3-A.

2.0 DETAILED DESCRIPTION

2.1 PROPOSED CHANGE

The proposed license amendment would revise TS Section 6.15 as shown below.

> 6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM
>
> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with ~~the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995," as modified by the following exception:~~ *Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A.*
>
> ~~a. NEI 94-01-1995, Section 9.2.3: The first ILRT performed after October 30, 1992 shall be performed no later than April 29, 2008.~~

A markup of the TS showing the proposed changes is provided in Attachment 2.

The purpose of NEI 94-01, Revision 3-A guidance is to assist licensees in the implementation of Option B to 10 CFR 50, Appendix J, "Leakage Rate Testing of Containment of Light Water Cooled Nuclear Power Plants," (hereafter referred to as Appendix J, Option B). Revision 2-A of NEI 94-01 (Reference 6.3) added guidance for extending containment integrated leak rate test (ILRT or Type A test) surveillance intervals beyond ten years, and Revision 3-A of NEI 94-01 adds guidance for extending containment isolation valve (Type C test) local leakage-rate test (LLRT) surveillance intervals beyond 60 months.

The technical basis for the proposed license amendment utilizes risk-informed analysis augmented with non-risk related considerations. A risk impact evaluation performed by Westinghouse Electric Company (WEC) concluded that the increases in large early release frequency (LERF) are within the limits set forth by the applicable guidance contained in NRC Regulatory Guide (RG) 1.174 (Reference 6.4), NUREG-1493, "Performance-Based Containment Leak-Test Program," and EPRI Technical Report TR-1009325 (Reference 6.5).

In accordance with the guidance of NEI 94-01 Revision 3-A, NextEra proposes to extend the maximum surveillance interval for the ILRT to no longer than 15 years from the last ILRT based on satisfactory performance history. The current interval is no longer than 10 years and would require that the next ILRT for Seabrook be performed during the Spring 2017 refueling outage. The proposed change would allow the Seabrook ILRT to be performed in 2023. This will reduce the number of ILRTs performed over the licensed period of operation resulting in significant savings in radiation exposure to personnel, cost, and critical path time during refueling outages.

2.2 DESCRIPTION OF SEABROOK PRIMARY CONTAINMENT

The Seabrook primary containment structure is a seismic Category I reinforced concrete dry structure, which is designed to function at atmospheric conditions. It consists of an upright cylinder topped with a hemispherical dome, supported on a reinforced concrete foundation mat which is keyed into the bedrock by the depression for the reactor pit and by continuous bearing around the periphery of the foundation mat. The inside diameter of the cylinder is 140 feet and the inside height from the top of the base mat to the apex of the dome is approximately 219 feet; the net free volume is approximately 2,704,000 cubic feet.

A welded steel liner plate, anchored to the inside face of the containment, serves as a leaktight membrane. Although not a code requirement, welds that are embedded in concrete and not readily accessible are covered by a leak chase system which permits leak testing of those welds throughout the life of the plant. Exemptions to these inaccessible welds are the welds joining mechanical penetrations X-60 and X-61 to the steel liner plate. (The venting pipes which join the leak chase channels for these penetrations to the atmosphere were not provided; however, these welds underwent proper testing before they became inaccessible.) The liner on top of the foundation mat is protected by a four feet thick concrete fill mat which supports the containment internals and forms the floor of the containment.

The containment is designed to assure that the base mat, cylinder, and dome behave integrally to resist all loads.

Located outside the containment building and having a similar geometry is the containment enclosure building. This structure provides leak protection for the containment and protects it from certain loads.

a. Base Mat

The reinforced concrete base mat is 153 feet in diameter and 10 feet thick. It is designed to carry the loads from the shell of the containment and from the internal structures.

An orthogonal grid rebar arrangement is provided for the bottom face of the base mat to simplify fabrication and construction. A radial and hoop pattern is used at the top face to minimize interference with cylinder dowels. Vertical and inclined shear reinforcement are provided to resist the transverse shear forces caused by design accident pressure and seismic loads. The mat liner plate is 1/4" thick with joints welded to leveling angles which serve as welding backup strips.

Internal structures are supported on and anchored to the fill mat, as indicated above. The mat is not anchored to the base mat. Stability of the containment with internals is provided by the keying action of the base mat and reactor pit in the rock and by bearing against the foundation for the Enclosure Building, which in

turn transfers all horizontal shears directly into the bedrock through fill concrete.

b. <u>Cylinder</u>

The cylinder has an inside diameter of 140 feet and is nominally 4'-6" thick. Also, it is thickened to provide room for additional reinforcing steel around the openings for the equipment hatch and the personnel air lock.

The reinforcing bars in the cylinder are arranged and oriented to resist hoop, meridional and shear forces, including hoop, meridional and radial shear forces produced by bending moments. Orthogonal layers of bars in the meridional and hoop directions are provided on each face to resist the membrane forces primarily from pressure and seismic loads.

An orthogonal set of bars inclined at 45 degrees to the horizontal is provided on the outside face to resist in-plane seismic shear forces and membrane tension from other loads. Near the base of the containment, additional meridional bars and radial inclined stirrups are provided to resist discontinuity moments and radial shears, respectively, caused by the restraint on the cylinder at the junction of the cylinder and the base mat. Stirrups are also provided at the springline to resist radial shear.

Where there are large openings for access hatchways and penetrations, the main reinforcing bars are continued without interruption around the openings. No main reinforcing bars are terminated at any opening. Furthermore, additional bars are provided to resist the local effects of these openings and, around large openings such as the equipment hatch (28'-0" inside diameter) and personnel air lock (7'-1¼" inside diameter), the concrete is thickened locally to resist the additional local forces and to accommodate the additional reinforcing.

The liner plate in the cylinder is 3/8" thick in all areas except penetrations and the junction of the base mat and cylinder where it is 3/4" thick. The liner is provided with an anchorage system to assure that it can withstand accident loadings while maintaining leak tightness. In addition, the anchorage system assures that the liner, which is used as a form during construction, can resist the hydrostatic concrete loads while maintaining liner tolerances within allowable values. The anchorage system consists of vertical tees spaced every 20 inches around the circumference of the cylindrical wall. The webs of the tees are welded to the liner plate with two ¼" continuous fillet welds. Bent studs are attached to the flange of vertical tees as required to accommodate placement of rebar.

Containment penetrations, other than the equipment hatch and personnel air lock, are located in the lower portion of the cylindrical structure. In general, a penetration consists of a sleeve anchored in the concrete cylinder wall and welded to the locally thickened containment liner. The weld between the liner and the sleeve is covered by a leak chase system which can be pressurized to demonstrate the integrity of the penetration-to-liner weld joint. The piping, electrical cable and instrumentation cable pass through the embedded sleeves and the ends of the resulting annuli are closed off either by welded end plates or a flued head welded to the sleeve outside the containment. If the pipe carries hot fluid, the space between the pipe and the sleeve is insulated to maintain the concrete temperature adjoining the embedded sleeve at or below 200° F during normal plant operation.

The fuel transfer tube passes through an embedded sleeve which has its ends closed off by an expansion bellows and an end plate. In the case of ventilation ducts, the sleeve forms the wall of the duct.

Sleeves for all penetrations, including the equipment hatch and personnel air lock, are embedded in the concrete wall by an engineered anchorage system that is welded to the penetration sleeve. Reinforcing steel, hoop, meridional and diagonal, is splayed around penetrations permitting all bars to be continuous.

All brackets and attachments are welded to attachment plates which are welded to the liner plate and anchored into the concrete by studs welded on the opposite side of the liner, thereby transmitting the forces directly to the concrete. In the reactor cavity pit, brackets are used, as a construction aid, on the concrete side of the liner to temporarily support the reinforcing steel. This is a permanent attachment which only functions until the concrete is placed. Each of these brackets is welded to an attachment plate; the attachment plate is welded to the liner with a continuous fillet weld and is not backed by a stud.

c. Dome

The dome is a reinforced concrete shell 3'-6" thick and 69'-11" in radius. Due to the change in concrete thickness, the discontinuity of concrete at the springline is on the outer surface.

Reinforcing steel in the dome consists of hoop, meridional, and diagonal bars, as in the cylinder. The meridional and diagonal bars are continuous with those in the cylinder.

One-half of the meridional bars, in an alternating fashion, are terminated at 60° above the springline with the remaining bars evenly spread and continued across the upper 30° of the dome.

The hoop bars in the dome are terminated where no longer needed at 75° above the springline.

One-half of the diagonal bars are terminated where no longer needed, at approximately 30° above the springline; the remaining bars are terminated at approximately 45° above the springline.

The dome liner is 1/2" thick and flush with the outside face of the cylindrical liner. The anchorage system consists of tees on a 5'-0" grid pattern. A bent stud is located in the center of each of the resulting 5'-0" x 5'-0" panels to provide some additional anchorage.

d. Steel Components

Steel components that resist pressure and are not backed by structural concrete include the following:

1. Equipment hatch
2. Personnel air lock
3. High energy piping penetrations
4. Moderate energy piping penetrations

5. Electrical penetrations
6. Fuel transfer tube assembly
7. Instrumentation penetrations
8. Ventilation penetrations

3.0 TECHNICAL EVALUATION

3.1 LEAK TEST HISTORY

3.1.1 Type A Testing

The historical results of the Type A tests for Seabrook are included in the table provided below. The reported leak rate is at the 95 percent upper confidence level and includes any Type B and Type C penalties.

The last Seabrook Type A test was completed on April 27, 2008. Previous Type A testing confirmed that the Seabrook containment structure leakage is acceptable with respect to the TS acceptance criterion of 0.15 percent of primary containment air weight per day at the design basis loss of coolant accident pressure (P_a). Since the last two Seabrook Type A test as-found results, as shown in the table provided below, were less than 1.0 L_a, a test frequency of at least once per 10 years is justified in accordance with NEI 94-01, Revision 0.

Repair or replacement activities (including any unplanned activities) performed on the pressure retaining boundary of the primary containment prior to the next scheduled Type A test would be subject to the leakage test requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Paragraph IWE-5221, "Leakage Test." There have been no pressure or temperature excursions in the containment that could have adversely affected containment integrity. There is no anticipated addition or removal of plant hardware within containment that could affect leak-tightness that would not be challenged by local leak rate testing.

Following the approval of this license amendment, the next Seabrook Type A test must be performed on or before April 27, 2023.

In the following table, the 1986 and 1989 ILRTs were performed using the Mass Point Method, whereas the 1992 and 2008 ILRTs were performed using Total Time. This accounts for the differences in the test results in the table below.

Seabrook Type A Test Historical Results Since 1986

Test Completion Date	Test Pressure (PSIA)	P_a (PSIG)	As Found Leak Rate	As Found Acceptance Criteria	As Left Leak Rate	As Left Acceptance Criteria
03/19/1986	64.923	49.6	0.058 %wt/day	≤ 0.15 %wt/day	0.058 %wt/day	≤ 0.1125 %wt/day
11/23/1989	64.849	49.6	0.0601 %wt/day	≤ 0.15 %wt/day	0.059 %wt/day	≤ 0.1125 %wt/day
10/30/1992	65.4	49.6	0.10748 %wt/day	≤ 0.15 %wt/day	0.10594 %wt/day	≤ 0.1125 %wt/day
4/27/2008	50.31 PSIG	49.6	0.1004 %wt/day	≤ 0.15 %wt/day	0.0999 %wt/day	≤ 0.1125 %wt/day

%wt/day = Percent primary containment air weight per day

The following is a description of the results from the latest two ILRTs at Seabrook.

The April 2008 periodic Type A test was performed using BN-TOP-1 calculated at the 95% upper confidence limit (UCL). The performance leak rate corresponding to the definition in NEI 94-01 was equal to the as-left ILRT results of 0.0999 %wt/day since no leakage paths were isolated during the ILRT.

The October 1992 periodic Type A test was performed using BN-TOP-1 calculated at the 95% UCL. The performance leak rate corresponding to the definition in NEI 94-01 was equal to the as-left ILRT results of 0.10594 %wt/day since no leakage paths were isolated during the ILRT.

As required by NEI 94-01, Revision 3-A Section 9.1.2, further extensions in test intervals are based upon two consecutive, periodic, successful Type A tests and requirements stated in Section 9.2.3 of this guideline. The results in the table show that there has been some margin to the maximum allowable leakage rate of 0.15 %wt/day.

3.1.2 Type B and C Testing

The Type B and Type C containment leakage rate testing program for Seabrook requires pneumatic tests intended to detect or measure leakage across pressure-retaining or leakage limiting boundaries and containment isolation valves. As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority of potential containment leakages.

As discussed in NUREG-1493 and NEI 94-01, Revision 3-A, Type B and Type C tests can identify the vast majority of all containment leakage paths. This amendment request adopts the guidance in NEI 94-01, Revision 3-A in place of NEI 94-01, Revision 0, but otherwise does not affect the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that containment leakage rates are maintained well within limits.

A review of the Type B and Type C test results from the spring of 2005 through the fall of 2015 has shown a large amount of margin between the actual as-found and as-left outage summations and the TS leakage rate acceptance criteria (that is, less than 0.6 L_a).

- The as-found minimum pathway leak rate for Seabrook shows an average of 8.3 percent of 0.6 L_a.
- The as-left maximum pathway leak rate for Seabrook shows an average of 14.6 percent of 0.6 L_a with a high of 18.1 percent or 0.109 L_a.

Seabrook Type B and Type C Leak Rate Summation History Since 2005

Refueling Outage	As-Found Min Path	Percentage of 0.6 L_a	As-Left Max Path	Percentage of 0.6 L_a
OR 10 Spring 2005	26.457 scfh	6.0%	58.911 scfh	13.3%
OR 11 Fall 2006	30.059 scfh	6.8%	58.600 scfh	13.3%
OR 12 Spring 2008	25.770 scfh	5.9%	52.946 scfh	12.0%

OR 13 Fall 2009	44.200 scfh	10.0%	73.253 scfh	16.6%
OR 14 Spring 2011	37.581 scfh	8.5%	68.184 scfh	15.4%
OR 15 Fall 2012	24.637 scfh	5.6%	55.817 scfh	12.6%
OR 16 Spring 2014	52.848 scfh	12.0%	68.618 scfh	15.5%
OR 17 Fall 2015	50.660 scfh	11.5%	80.233 scfh	18.1%

scfh = standard cubic feet per hour

There has been one local leak rate test failure in the past 36 months. During OR15, check valve IA-V-531 failed the local leak rate test.

The inside containment isolation check valve for instrument air penetration X-68 is IA-V-531. This valve is 2" BW/IP (Kerotest) Y pattern check valve with ethylene propylene terpolymer (EPT) seats. The valve is a Critical 1, Safety Class 2, Seismic 1, IST, and included in the Appendix J program. This valve was tested under Appendix J on an extended frequency based on acceptable past performances. This valve was replaced in March 1999 during OR06 due to increasing leak rate. The existing valve was replaced with the soft seated check valve now in service. Local leak rate testing was performed as a post maintenance test in OR06 with a 0.279 scfh test result. The local leak rate test was performed the next two outages to establish a performance base prior to any interval extensions. The test results for OR07 and OR08 were 0.542 and 0.589 scfh respectively. The extended frequency was applied and the valve was tested in OR11 with local leak rate result of 1.492 scfh. The valve was tested again in OR14 with test result of 4.592 scfh. Condition reports 01673034/01682493 were initiated to address the increasing trend in the local leak rate results after OR14 for a soft seat replacement in OR15. Work order 40132878 was issued to perform the soft seat replacement in OR15. The as-found local leak rate results were found to be gross (not able to hold pressure). Note the outside containment boundary was satisfactory. The existing work order was used to disassemble and inspect the valve. The as-left local leak rate results were satisfactory at 3.569 scfh.

The outside containment fail closed AOV for instrument air penetration X-68, IA-V-530, has maintained a consistently low LLRT result.

Discussion: IA-V-531 was disassembled, inspected, soft seats replaced, reassembled and retested. The valve disassembly was not conclusive in determining the cause of the local leak rate failure. Discussions with the mechanics and the write up in the work order did not indicate any excessive debris or grit. The initial cleaning of the valve was done by a contractor and he did not discuss with an in-house mechanic if there was any grit or debris in the valve. The installed soft seat was examined and found to be harder (not as pliable as a new soft seat) though no hardness readings were obtained. The disc contact area on the seat was examined and found to be in overall good condition with no evidence of linear indications, pitting, or rolled edges. Note there was some evidence of slight indications (i.e. nicks and machining cuts) just outside of the actual machined main seating surfaces which should not impact the valve seating. The internal seat and external disc/plug dimensions were provided to Flowserve Corporation for tolerance verification. Flowserve Corporation validated that all dimensions are within tolerance.

This is a carbon steel system which typically would have some grit or debris. This valve is a Y pattern check valve with a disc and disc guide. This type of valve configuration has close tolerances between the disc/disc guide and the bore. Any grit or debris could potentially hang the disc/disc guide in the open position. Additionally, any debris under the seat with a hardened soft seat may allow the disc to not fully seat.

The check valve was flowed in the open direction before conducting the AS FOUND LLRT. This is a required test to be done even though this valve has no required OPEN safety function. The forward flow test is done using the LLRT Panel and clean filtered air.

The AS FOUND LLRT was then done and found to be excessive - could not pressurize the system. After documenting the failure, percussion adjustments were done to see if the disc would seat. These adjustments were not successful, indicating the valve was off the seat.

The age of the replacement seat may help explain why the AS LEFT test was non zero. A durometer test was not done, but the new soft seats felt slightly more pliable than the old soft seat.

This valve might have to be maintained on a one refueling outage frequency especially if the system does not have any filters in the line to containment.

Conclusion: Although there was no conclusive determination of the cause of IA-V- 531 not seating mentioned in the OR15 repair work order, the most probable cause of IA-V-531 not seating is debris/grit in the seat with the hardened soft seat insert or in the check valve bore.

The local leak rate test was performed satisfactorily in OR16 and OR17, which re-establishes valve performance.

A new soft seat insert has been installed. The valve internals have been cleaned. The AS LEFT LLRT was SAT.

There are performance factors that need to be considered before applying an extended testing interval. These performance factors have been evaluated and documented at Seabrook as follows:

For purposes of determining an extended test interval, an assessment of the containment penetration and valve performance should be performed and documented. The following items should be considered in establishing and implementing extended test intervals for Type B and C components:

- Past Component Performance - Specific component performance of two successful consecutive as-found Type B or C tests are performed.
- Service - The environment and use of components in determining their likelihood of failure based on their performance history.
- Design - Valve type and penetration design may contribute to their leakage characteristics.
- Safety Impact - The relative importance of penetrations due to the potential impact of failure in limiting releases from containment under accident conditions.
- Cause Determination - For failures identified during an extended test interval, a cause determination should be conducted and appropriate corrective actions identified to address common-mode failure mechanisms.

For Type B testing, 3 of 13 penetrations are currently on extended frequency. Two of the 13 penetrations are electrical penetrations. Since the electrical penetrations are train related, each penetration is tested every other refueling outage. Penetrations on extended frequency have performed well. Measured leakage has not changed significantly over 120 months.

For Type C testing, 26 of 37 eligible penetrations are on extended frequency. This does not include the two penetrations that are required to be tested on a 30 month frequency per Regulatory Guide 1.163. Of the 11 penetrations not on extended frequency, four are train related and are tested every other outage. These four penetrations would have the potential to be moved out to 75 months (every four refueling outages) with amendment approval, without affecting the train-required testing schedule, since they have had acceptable performance history. One penetration is not on extended frequency due to the failure previously discussed. The remaining six penetrations are tested every outage. The percentage of penetrations on extended frequency is 70.3% and represents good performance, which supports allowing an extended test interval of up to 75 months for Type C tested CIVs in accordance with the guidance of NEI 94-01, Revision 3-A.

Table 1 – Extended Frequency Percentages

Group	Number of penetrations	Number of penetrations on extended frequency	% extended	Comment
Type B Penetrations	13	3	23.08%	Extended frequency is 120 months
Type C Penetrations	39	26	66.67%	Extended frequency is 60 months
Eligible Type C Penetrations	37	26	70.27%	Extended frequency is 60 months. Does not include components restricted by RG 1.163 paragraph C.2

Table 2 – Type B Penetrations most recent two tests

Penetration	Limit (scfh)	Most Recent Test	Leakage (scfh)	Previous Test	Leakage (scfh)
HVAC 1 Flange	147.8	OR17	0	OR16	0
HVAC 2 Flange	147.8	OR17	0	OR16	0.041
Fuel Transfer Tube Flange	147.8	OR17	0.076	OR16	0
Fuel Transfer Tube Bellows	147.8	OR16	0.292	OR13	0
Equipment Hatch Barrel	37	OR17	3.688	OR16	0
Equipment Hatch Seals	147.8	OR17	0.04	OR16	0
Personnel Hatch Barrel	37	10/2013	0.482	8/2011	0.535
Spare (E-58)	147.8	OR17	0	OR16	0
Spare (E-59)	147.8	OR17	0	OR15	0
H2 Analyzer Train A	147.8	3/2008	1.173	6/2002	1.242

H2 Analyzer Train B	147.8	10/2014	1.755	1/2011	0.948
Electrical Penetrations 1	147.8	OR17	0.041	OR15	0.042
Electrical Penetrations 2	147.8	OR16	0.176	OR14	0.151

Leakage units are scfh – standard cubic feet per hour

Table 3 – Type C Penetrations most recent two tests

Penetration	Valves	Limit	Most Recent Test	Leakage (scfh)	Previous Test	Leakage (scfh)
X-09	**RC-V23 and RC-V24**	147.8	OR17	0.525	OR15	0.510
X-10	**RC-V88 and RC-V89**	147.8	OR16	15.925	OR14	5.624
X-14	**CBS-V12/CBS-V11**	147.8	OR17	2.191	OR15	1.844
X-15	**CBS-V18/CBS-V17**	147.8	OR16	3.576	OR14	1.742
X-16	**COP-V3 and COP-V4**	7.4	OR17	1.638	OR16	1.603
X-17	**VG-FV1712/VG-FV1661**	147.8	OR15	0	OR12	0
X-18	**COP-V2 and COP-V1**	7.4	OR17	4.348	OR16	4.623
X-19	**SS-V273/SS-FV2857**	147.8	OR15	1.020	OR12	0.289
X-20	**CC-V168/CC-V57 and CC-V-845**	147.8	OR16	0	OR13	3.059
X-21	**CC-V122/CC-V121 and CC-V-410**	147.8	OR17	0.292	OR14	0.903
X-22	**CC-V257/CC-V256 and CC-V474**	147.8	OR15	0.612	OR12	0.287
X-23	**CC-V175/CC-V176 and CC-V840**	147.8	OR15	3.058	OR12	7.163
X-32	**WLD-V82/WLD-V81 and WLD-V213**	147.8	OR17	0	OR14	1.944
X-34	**WLD-FV8330/WLD-FV8331 and WLD-V209**	147.8	OR16	0	OR15	0
X-35A	**SI-V62 and SI-V157/SI-V70 and SI-V247**	147.8	OR17	0	OR14	1.334
X-35B	**RC-FV2840/RC-FV2830 and RC-FV2831 and RC-V312**	147.8	OR17	5.393	OR14	2.453
X-35C	**RC-FV2874 and RC-FV2894/RC-FV2832 and RC-V314**	147.8	OR17	3.385	OR16	4.884
X-35D	**RC-FV2876 and RC-FV2896/RC-FV2833 and RC-V337**	147.8	OR17	0.331	OR16	0.717
X-36A	**DM-V4/DM-V5 and DM-V18**	147.8	OR15	0.408	OR12	0
X-36B	**NG-V13/NG-V14**	147.8	OR16	0.618	OR13	0
X-36C	**RMW-V29/RMW-V30**	147.8	OR15	0	OR12	0.409
X-37A	**CS-V149/CS-V150**	147.8	OR16	0	OR15	0
X-37B	**CS-V167/CS-V168 and CS-V794**	147.8	OR15	0	OR13	0
X-38A	**FP-V588/FP-V592**	147.8	OR17	11.10	OR16	32.86
X-38B	**CGC-V46/CGC-V43 and CGC-V44 and CGC-V45**	147.8	OR17	3.049	OR16	2.265
X-39	**SF-V87/SF-V86 and SF-V101**	147.8	OR17	0	OR15	0
X-40A	**NG-FV4609/NG-FV4610**	147.8	OR15	0	OR12	0
X-40B	**RC-FV2836/RC-FV2837**	147.8	OR15	0	OR12	0
X-52A	**CAH-FV6572/CAH-FV6573**	147.8	OR15	0.286	OR12	0
X-52B	**CAH-V12/CAH-FV6574**	147.8	OR17	0.240	OR14	0.287

X-67	**SA-V229/SA-V1042**	147.8	OR17	0.269	OR14	0.411
X-68	**IA-V530/IA-V531**	147.8	OR17	19.661	OR16	10.661
X-71A	**CGC-V32/CGC-V34**	147.8	OR15	0	OR12	0.515
X-71B	**CGC-V24/CGC-V25**	147.8	OR17	3.563	OR14	0.820
X-71C	**CGC-V28/CGC-V36**	147.8	OR15	0.614	OR12	0.816
X-71D	**LD-V1/LD-V2**	147.8	OR15	0.285	OR12	0
X-72A	**CGC-V10/CGC-V12**	147.8	OR15	0	OR12	0
X-72B	**CGC-V3/CGC-V4**	147.8	OR17	0.722	OR14	0.288
X-72C	**CGC-V14/CGC-V15**	147.8	OR16	0	OR13	0

Leakage units are scfh – standard cubic feet per hour

3.2 CONTAINMENT INSPECTIONS

General visual examinations of the accessible surfaces of the primary containment are performed in accordance with the Containment Inservice Inspection Program. These examinations are performed to assess the general structural condition of the containment building reinforced concrete and to satisfy the visual examination requirements of ASME Code Section XI, Subsection IWL. These examinations are performed in sufficient detail to identify areas of concrete deterioration and distress.

Detailed visual examinations are performed to determine the magnitude and extent of deterioration of suspect surfaces initially detected by general visual examinations. The conditions reported during the examinations are evaluated to determine acceptability. The conditions are acceptable if it is determined that there is no evidence of damage or degradation sufficient to warrant further evaluation or performance of repair and replacement activities. These concrete examinations are performed on a five year frequency.

The metal containment liner is visually examined under two separate programs. The first is the Containment Inservice Inspection Program discussed in Section 3.2.1. This program includes provisions to satisfy the visual examination requirements of ASME Code Section XI, Subsection IWE and 10 CFR 50, Appendix J, Option B. A visual examination is made of the accessible interior surfaces of containment in order to identify evidence of deterioration that may affect the containment structural integrity or leak tightness. If signs of corrosion are evident that exceed the acceptance standard (IWE-3500), they must be either corrected by a repair or replacement activity or deemed acceptable for continued service by an engineering evaluation. Both Regulatory Guide 1.163, September 1995, and the ASME Code require a general visual examination of the accessible liner surfaces three times in a ten year period.

The second program is the Containment Coatings Inspection and Assessment Program discussed in Section 3.2.6. This program mandates a visual inspection and assessment of the protective coatings on the containment structure and equipment in the readily accessible areas of the reactor containment building. The examination areas are selected such that all painted surfaces are inspected over a two outage time frame. This program is implemented to ensure that the integrity of the coatings is maintained and was established in response to NRC Generic Letter 2004-02.

The inspection frequency of the above programs ensures that when an area of concern is identified, it only affects a small localized area. Corrective action is taken following any signs of paint blistering, peeling, or corrosion.

3.2.1 Containment Inservice Inspection Program

The requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section III, Division 2, Code for Concrete Reactor Vessels and Containments were used in the design and construction of the Seabrook containment structures. The Concrete Containment Component was designed to the 1975 Edition and the Containment Liner was designed to 1975 Edition (1976 Winter Addendum used for liner allowable stresses).

For the purposes of this plan, the Seabrook containment building is classified as a Class CC component with a metallic liner.

The Containment Inservice Inspection (ISI) Program applies to the containment vessel (ASME Code Section XI, Subsection IWE) and the containment reinforced concrete (ASME Code Section XI, Subsection IWL).

ASME Code Section XI, Subsection IWE specifies that examinations will be performed on the pressure retaining boundary of the containment vessel, which includes the accessible surfaces of the liner plate, integral attachments and structures that are part of the reinforcing structure, surfaces of pressure retaining welds, pressure retaining bolted connections, and the moisture barrier, which prevents moisture intrusion at the concrete-to-metal interface at the basement floor. Also, the containment surfaces that may require augmented examination are included in this program.

ASME Code Section XI, Subsection IWL specifies that examinations will be performed on the surfaces of the reinforced concrete of the containment building. This program does not include Item Numbers E1.12 (Wetted Surfaces of Submerged Areas), E1.20 (BWR Vent System), and Examination Category L-B (Unbounded Post Tensioning System). These circumstances and components do not exist at Seabrook.

In accordance with the NRC final rule amending 10 CFR 50.55a that was effective September 9, 1996, the Containment Inservice Inspection Plan (IWE and IWL) was developed with an initial interval start date of August 19, 2000. The 1995 Edition, 1996 Addenda of ASME Code Section XI was the basis for the programs. For ease of administration, the Subsection IWE and IWL interval dates and ASME Code Section XI Code Editions are synchronized with the remainder of the ASME Code Section XI subsections. The subsequent interval start date for the Subsection IWE and IWL programs was August 19, 2010 at Seabrook; the subsequent interval code of record is ASME Code Section XI, 2004 Edition, no Addenda, and implemented in accordance with the Seabrook Station Reference Manual – Inservice Inspection Reference (SIIR).

<u>Inspection Interval and Inspection Periods</u>

The required Subsection IWE and IWL examinations are scheduled and tracked using a database. The current and the next containment in-service inspection intervals for Seabrook are summarized in the tables below:

Current IWE/IWL Interval

System Identification	Examination Description	Item Number	Exam Method	Period Scheduled		
Examination Category E-A				1	2	3
Containment Liner	Accessible Surface Areas	E1.11	GV	1	1	1
	Pressure Retaining Bolting	E1.11B	GV	1	1	1
	Pressure Retaining Bolting	E1.11C	VT-3 Note 1			
	Moisture barrier	E1.30	GV	1	1	1
Examination Category E-C				1	2	3
	Visible Surfaces	E4.11	VT-1 Note 2			
	Surface Area Grid Minimum Wall Thickness Location	E4.12	UTT			
Examination Category L-A					5 year inspection	
	Concrete Surface – All Accessible Surface Areas	L1.11	GV		2015 Note 3	
	Concrete Surface – Suspect Areas	L1.12	GV			

Item Number — Refers to item numbers listed in ASME Code Section XI, Table IWE-2500-1, titled "Examination Categories"

Exam Method — GV - General Visual;
UTT- Ultrasonic Thickness Test; and
VT -3- examination method defined in ASME Code Section XI, Paragraph IWA-2213, "VT -3 Examination"
VT-1 – examination method defined in ASME Code Section XI, Paragraph IWA 2211, “VT-1 Examination”,
Note 1: An examination of the pressure-retaining bolted connections in Item E1.1 of Table IWE-2500-1 using the VT-3 examination method must be conducted once each interval per 10CFR50.55a(b)(2))(ix)(G). Per 10CFR50.55a(b)(2))(ix)(H), containment bolting that is disassembled during scheduled performance of the examinations in Item E1.11 of Table IWE-2500-1 must be examined using the VT-3 method.
Note 2: required per 10CFR50.55a(b)(2)(ix)(G)
Note 3: Examination will be completed in 2016 as allowed by IWL-2400(c).

Schedule — The scheduled dates or refueling outages for inspections during the current Containment Inservice Inspection Interval are based on requirements of ASME Code Section XI, Tables IWE-2500-1, and Table IWL-2500-1.

Containment Surfaces Subject To Augmented Examinations

Seabrook has no areas subject to ASME Code Section XI, Subsection IWE, augmented examinations.

3.2.1.1 Containment IWE Inspections

Recent Containment Liner Accessible Surface Inspection (E1.11)

The containment liner accessible surface areas were inspected in OR17 in the fall of 2015. The following deficiencies were noted:

During OR17, IWE examination identified 35 areas of degradation on the containment liner surface areas.

Additionally, 59 areas of degradation resulting from the ASME Section XI, IWE examinations in previous outages that were not remediated during OR17 were reviewed.

An Engineering Evaluation, per IWE-3222(3) determined that the areas of degradation on the metal containment liner, containment penetration sleeves, and containment hatches that were not remediated during OR17 were nonstructural in nature and had no unacceptable effects on the structural integrity of the containment. These areas have degraded coatings (including chips, scratches, blisters, staining or discoloration in the coating film) with evidence of corrosion.

Remediation of these indications was deferred for 18 months until OR18 based on the following;

- These indications (chips, scratches, blisters) are small and typically range in size from ⅛ inch to ½ inch in length and less than 1 sq. inch of effected area of the containment liner. The surrounding coating material is tightly adhered to the metal substrate and there are no other signs of distress in the surrounding coating material. These indications are randomly scattered on the containment liner and are not interconnected. These indications will not impact the Seabrook debris generation calculation or the inventory of unqualified coatings in containment calculation.

- These indications were caused by mechanical damage or impact to the metal containment liner. These indications were not caused by corrosion of the metal containment liner under the coating material.

- The reported staining and or discoloration of coating was typically debris or deposits of foreign material such as residual duct tape adhesion, grease, and airborne deposits on the surface of the coating film. None of these deposits will adversely affect the integrity of the coating material or the adhesion of the coating to the metal substrate.

- The corrosion is typically limited to surface oxidation which can be removed by either hand or power tool cleaning techniques. There was no evidence of pitting or metal loss below the nominal wall thicknesses

- Exposure of the bare uncoated metal surfaces to the containment atmosphere during the next 18 month operating cycle will not produce any appreciable corrosion or any pitting in the bare exposed metal substrates.

Based on the above, it was concluded that these indications or imperfections in the coating

material are nonstructural in nature and have no effect on the design function of the metal containment liner or structural integrity of the containment structure.

IWE-2420 requires successive inspection of areas accepted by Engineering Evaluation. IWE-2420 requires that reexamination be performed during the next period. The IWE program has been adjusted to reflect this requirement. These areas will be reexamined in OR18.

Conclusion

Seabrook containment liner accessible surface inspections were successfully completed during OR17. The Seabrook containment liner surface continues to remain capable of performing its safety-related function. Successive inspections will be performed on areas of degradation accepted by evaluation in OR18.

Recent Containment Moisture Barrier Inspection (E1.30)

This section discusses the recent inspection of the containment liner wall to concrete floor moisture barrier. A detailed discussion of the containment leak chase channel plug moisture barrier is provided in section 3.2.3.

The containment liner wall to concrete floor moisture barrier was inspected in OR17 in the fall of 2015. The following deficiencies were noted:

- Cracked, chipped, blistered floor coatings and cracks in the concrete adjacent to the moisture barrier.
- Degraded coatings and corrosion particles on the metal containment liner adjacent to the moisture barrier.
- Separations between the moisture barrier and the concrete floor slab and separations between the metal containment liner and the moisture barrier.
- Lack of or insufficient moisture barrier material.

These deficiencies compromise the design function of the moisture barrier to seal the joint between the metal containment liner and the concrete floor slab.
Repairs were performed on all degraded areas to include:

- All loose and heavily degraded floor coatings were removed. Coatings were repaired as part of the Service Level 1 coating program.
- All cracks and spalling the concrete that could affect the moisture barrier seal were filled with an approved epoxy filler compound.
- All degraded coatings and corrosion products were removed from the metal containment liner. UT measurements were taken to verify nominal wall thickness. The areas were then recoated with a Service Level 1 coating. All UT measurements were above the nominal wall thickness.
- Degraded areas of the moisture barrier itself were repaired.

Conclusion

Seabrook moisture barrier inspections and repairs were successfully completed during OR17. The Seabrook containment building containment liner wall to concrete floor moisture barrier continues to remain capable of performing its safety-related functions.

3.2.1.2 Containment IWL Inspections

<u>Recent Containment Concrete Surface Inspection (L1.11)</u>

The containment concrete surface was inspected in the fall of 2010. Eighty-four (84) suspect areas were identified that required an Engineering Evaluation. The discontinuities were categorized as; popouts, voids, hairline cracks < 40 mils, scaling, spalling and drummy areas. One crack was identified with a width greater than 40 mils and a limited length of 1 inch. These discontinuities were evaluated as acceptable without remediation based on the following:

- The discontinuities are located in the cover concrete with no exposed reinforcing steel.
- The affected area of concrete and the area surrounding the discontinuities is sound, durable concrete.
- There are no other signs of concrete degradation as noted on the General Visual examination reports.
- These discontinuities are passive and due to the benign mild service environment in the area, will not propagate or cause progressive degradation.
- These discontinuities will have no detrimental effect on the durability of the containment concrete or adversely impact the structural or functional integrity of the containment structure.

During walkdown assessments of the containment as part of the Structural Monitoring Program, four isolated locations were identified where the concrete had pattern cracking, which is typical of Alkali-Silica Reaction (ASR). Three locations were at the lower elevations where water in ingress into the Containment Enclosure Building (CEB) led to direct exposure of the outer surface of containment to water. The fourth location was in the Mechanical Penetration Area which did include evidence of water intrusion at the seismic isolation joint. At one location, the Combined Cracking Index (CCI) met the action level criterion in the Structural Monitoring Program, necessitating a structural evaluation. Seabrook has committed as part of License Renewal to maintain the exterior surface of the Containment Structure, from elevation -30 feet to +20 feet, in a dewatered state (see section for 3.2.7 for additional information).

In the fall of 2012, a structural evaluation was performed. This evaluation and the associated Prompt Operability Determination (POD), determined that the containment is capable of meeting all its design basis functions with the observed pattern cracking. Continued monitoring per the Structural Monitoring Program will identify any further expansion attributed to ASR.

Conclusion

Seabrook concrete surface inspections were successfully completed in 2010. Evaluations have been performed for observed ASR to demonstrate the containment building structural integrity. The Seabrook containment building concrete surface continues to remain capable of performing its safety-related function. In addition to being capable of providing the structural capacity of containment, there is also no evidence of impact to the containment liner as a result of ASR. See Section 3.2.4 for additional discussion on containment liner corrosion.

Seabrook will be performing a Containment Concrete Surface Inspection in the fall of 2016.

3.2.2 Containment Visual Inspection

Test Description

Containment and containment enclosure surface inspection procedure, EX1803.004, for Seabrook is utilized to perform general visual observations of the accessible interior and exterior surfaces of the containment structure in order to identify evidence of deterioration that may affect the containment structural integrity or leak tightness in accordance with the following.

• Technical Specification Surveillance Requirement 3.6.1.6 requires, in part, visual examinations in accordance with the Containment Leak Rate Testing Program.

• Technical Specification 6.15 requires, in part, visual examinations in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995. (Regulatory Position 3 requires that these examinations should be conducted prior to initiating a Type A test and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration.)

With the implementation of the proposed change, Technical Specification 6.15 will be revised by replacing the reference to Regulatory Guide 1.163 with reference to NEI 94-01, Revision 3-A. A general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity is required by NEI 94-01, Revision 3-A, prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years.

Recent Examination Results

The following is a summary of results of examinations that were recently performed.

An examination was successfully completed during the spring 2014 refueling outage (OR16) for both the containment interior steel liner surfaces and containment exterior concrete surfaces of the Seabrook containment building. The conditions identified were minor in nature and would not affect the integrity of the containment. Identified conditions were documented in the inspection report and do not require action.

Conclusion

Seabrook containment visual inspections were successfully completed during OR16. The Seabrook containment building continues to remain capable of performing its safety-related functions.

Supplemental Inspection Requirement

With the implementation of the proposed change, Technical Specification 6.15 will be revised by replacing the reference to Regulatory Guide 1.163 with reference to NEI 94-01, Revision 3-A. This will require that a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years.

The performance of containment structural integrity tests is utilized to perform general visual observations of the accessible interior and exterior surfaces of the containment structures. These examinations are also utilized to perform inspections in accordance with ASME Code Section XI, Subsections IWE and IWL. Additionally, the tests will continue to be performed to

meet the requirements of Technical Specification 6.15 with the incorporation of NEI 94-01, Revision 3-A guidelines.

3.2.3 Containment Liner Test Channel Plugs

The U.S. Nuclear Regulatory Commission (NRC) issued information notice (IN) 2014-07 to inform addressees of issues identified by the NRC staff concerning degradation of floor weld leak-chase channel systems of steel containment shell and concrete containment metallic liner that could affect leak-tightness and aging management of containment structures.

The containment basemat metallic shell and liner plate seam welds of pressurized water reactors are embedded in 3- to 4-feet (0.9- to 1.2-meter) thick concrete floor during construction and are typically covered by a leak-chase channel system that incorporates pressurizing test connections. This system allows for pressure testing of the seam welds for leak-tightness during construction and also in service, as required. The concrete floor thickness at Seabrook is nominally 4 feet.

In OR17 (fall 2015) examinations of the leak chase test connections were performed to address Information Notice 2014-07, Degradation of Leak Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner and to comply with the requirements of the IWE program.

At Seabrook there are 57 leak chase test connections. Eighteen of the leak chase test connections (LCTC) were not examined as they were either inaccessible or the outer (floor) cover could not be removed. All accessible LCTC floor covers were removed to inspect the condition of the port.

NextEra evaluated the conditions found during the inspections and concluded that the inaccessible portions of the liner plate have maintained and will continue to maintain their specified design function. During the next refueling outage, NextEra plans to perform further examinations to verify liner plate integrity and implement actions to remediate any remaining moisture issues.

No UT was possible due to the inadequate surface condition. Specifically, the areas were covered in debris and corrosion products that inhibited the adequate coupling of the transducer.

No other quantitative examinations were able to be performed. Qualified visual examinations were unsuccessful as well.

During examination of the leak chase test connections required by IWE-2500-1, Category E-A, Item E1.30 as moisture barriers, evidence of degradation to the moisture barrier was identified. Twenty seven locations were noted as having evidence of moisture/degradation inside the outer cover. Five locations were identified as having water present when the inner cover was removed. Visual examinations could not be performed past the inspection port pipe on eight locations due to stuck inner plugs. Twenty leak chase test connections were selected for further examination by video probe. Eleven leak chase test connections were limited due to the configuration of the tubing and did not include video into the chase (90 degree elbows in tubing).

The video probe inspections show the vertical sections of the risers were in generally good condition with corrosion of the side walls visible. Generally, more corrosion of the vertical risers was observed at the entrance, couplings, and nearest the entrance to the chases. No holes in the riser piping were observed. In most of the risers that contained 90 degree elbows, corrosion

product debris was visible at the bottom of the vertical section in the elbow. The risers that entered the leak chase allowed for video to be obtained of the chase in most cases. The chase was observed to be either damp, contain some water, completely filled with water, or dry. The chases observed typically contained some amount of corrosion product debris, regardless of whether the chase was damp, dry, or contained water.

Overall, the locations examined appear to be in good condition with no significant level of corrosion/metal loss identified.

Water samples were taken from various locations both inside and outside of the Unit 1 containment in an attempt to identify the source of water observed in leak chase observation ports opened during October 2015. This included five leakage chase channels where water was present when the inner cover was removed. However, the source water could not be definitively determined due to; consistent significantly elevated pH values obtained from all leak chase samples, and consistent sodium, potassium, chloride and sulfate values which differed from both plant system and groundwater concentrations, which indicated the water had been in contact with concrete for a long period of time.

The sample results from the leak chase locations had consistent elevated pH values between 12.5 to 13.1 and iron concentration values between 2 to 5 ppm, indicating the water was stagnant and in direct contact with concrete for a lengthy time period, likely several years. Based on this information, the steel liner and chase surfaces are protected with a tight, adherent iron hydroxide $Fe(OH)_2$ corrosion layer (or film) that is in equilibrium with the water above the metal surface. In addition, all leak chase samples chloride and sulfate concentrations were less than several hundred ppm indicating the iron hydroxide passive layer has not been disrupted. This means that the base metal has not been adversely impacted and is in a stable condition. Based on the sample pH values alone, the estimated corrosion rate is 0.0025 inches / year or 0.075 inches for a 30 year exposure. This corrosion rate is indicative of the stagnant water conditions in the leak chases.

The chemistry sampling was not able to conclusively identify the source of the leakage that was observed in the leakage chase channels. The most likely source is water that has seeped into the containment interior concrete floor over time and then leaks into the leak chase channel system through the channel welds, the seal welds on elbows, the piping weld to the leak chase channel, or through wall leakage in the piping or channel. Water that is stored in the concrete provides a low pressure water source (static head of water in the containment concrete) that is observed in the channel when the inner plug is removed.

ASME Section XI Subsection IWE paragraph IWE-3122.3 for acceptance of examination results by engineering evaluation that accepts conditions for continued service without repair or replacement requires that the evaluation indicates that the area of degradation is nonstructural in nature or has no unacceptable effect on the structural integrity of the containment. This evaluation indicates a minimum liner wall thickness of 1/8" is acceptable. The estimate corrosion rate based on the observed chemistry from the water samples is 0.075" over 30 years and the liner wall thickness is greater than the minimum. Therefore, there is no unacceptable effect on the structural integrity of the containment. Based on this evaluation, it is evident that the inaccessible portions of the liner plate have maintained their specified design function as a leaktight membrane.

Although a qualified NDE method could not be performed on the liner plate in locations associated with degraded moisture barriers, the evaluation supports the conclusion that the

inaccessible portions of the liner plate have not experienced any appreciable loss of material that would compromise the liner plate's ability to perform its specified design function.

The qualitative visual inspections performed did not show any significant degradation of the liner plate.

In summary, based on the examinations, chemistry analysis, design, and the corrosion assessment, reasonable assurance exists to conclude that the liner has maintained the ability to perform its specified design function and that the liner plate is capable of performing its specified design function for the next operating cycle.

The following actions will be completed no later than OR18 (Spring 2017):

- Complete the IWE Moisture barrier inspections of leak chase channels in OR18 for those penetrations where the outer cover or inner plug could not be removed.

  Moisture barriers are required to be examined under Table IWE-2500-1, Category E-A, Item No. E1.30. 100% of the moisture barriers are required to have a general visual examination performed once an Inspection period. Barriers not inspected in OR17 must be inspected in OR18. This will require the removal of the stuck outer covers and stuck inner plugs. This is required to complete the code required inspections. Eight leak chase channel systems were not inspected due to stuck inner plugs but had evidence of degradation. Nine leak chase channel systems and moisture barriers were not inspected due to stuck outer covers. Four moisture barriers were not inspected but there was no indication of a stuck outer cover. Five moisture barriers were determined to be inaccessible.

  Eighteen total moisture barriers remain to be inspected. Eight leak chase channel systems require inspection as well as any of the eighteen moisture barriers that are degraded.

- Evaluate those areas that were determined to be inaccessible to determine if the barrier for inaccessibility can be at least temporarily relocated to allow examination.

  10CFR50.55a(b)(2)(ix)(A)(1) requires the license to evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or could result in degradation to such inaccessible areas. The observed degradation in the accessible areas clearly indicates the likelihood of degradation in the inaccessible moisture barriers. Additionally, it is the responsibility of the Owner to maintain the access necessary to perform containment inspections when plant modifications are made. Therefore, it is reasonable to expect the Owner to consider actions that could make the moisture barriers accessible for examination.

3.2.4 Containment Liner Corrosion

The NRC has issued several information notices concerning containment corrosion, and NextEra reviewed these notices to determine the impact on the Seabrook containment. Several instances have been cited where organic material that is left lodged between the containment and the surrounding has resulted in through-wall or significant corrosion. Other instances cited focus on the moisture barrier. These events have been evaluated and inspections have been conducted to determine the presence of these conditions at Seabrook.

Section 5.1.5.1 of EPRI TR-1009325, Revision 2-A uses the Calvert Cliffs methodology in evaluating the impact of liner corrosion on the extension of ILRT testing intervals. This assessment was based on two observed corrosion events at North Anna Power Station, Unit 2 and Brunswick Steam Electric Plant, Unit 2. As there have been additional instances of liner corrosion in the industry that could be relevant to this assessment, a more complete accounting of all observed corrosion events relevant to Seabrook containment has been performed. Additionally, all relevant corrosion events are considered in the plant-specific risk assessment (Attachment 4) in support of this change.

NextEra reviewed industry information to identify corrosion events since 2000 (other than the North Anna Power Station Unit 2 and Brunswick Steam Electric Plant Unit 2 events) that are relevant to Seabrook. The March 2001 corrosion event at D.C. Cook discussed in NRC Information Notice (IN) 2004-09, Corrosion of Steel Containment and Containment Liner (Reference 6.6); and the April 2009 event at Beaver Valley Power Station Unit 1 discussed in IN 2010-12, Containment Liner Corrosion (Reference 6.7), are relevant corrosion events that resulted in through-wall holes. In March 2006 Beaver Valley Power Station Unit 1 discovered significant corrosion during the creation of a temporary opening in containment. This is also considered a relevant corrosion event.

Operating experience shows significant containment liner corrosion is often the result of liner plates being in contact with objects and materials that are lodged between or embedded in the containment concrete. Liner locations that are in contact with objects made of an organic material are susceptible to accelerated corrosion because organic materials can trap water that combined with oxygen will promote carbon steel corrosion. Organic materials can also cause a localized low pH area when they decompose. Organic materials located inside containment can come in contact with the containment liner and cause accelerated corrosion. The objects and materials that caused liner corrosion that licensees have found lodged between or embedded in the containment concrete include both foreign material (e.g., wooden pieces, workers' gloves, wire brush handles) and material that was deliberately installed as part of the design such as the felt material at Brunswick Steam Electric Plant, Unit 1. Inspections at Seabrook have detected no significant corrosion to the containment liner.

Corrosion at the containment liner to concrete floor interface, moisture barrier region, has resulted in limited corrosion. Coating failures have resulted in limited or general corrosion. These are not considered relevant to containment leakage. Additional detail on recent inspections is provided in Section 3.6.1.

3.2.5 Inaccessible Areas

For Class CC and MC applications, Seabrook shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, Seabrook shall provide the following in the ISI Summary Report, as required by 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A):

- A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
- An evaluation of each area, and the result of the evaluation, and;
- A description of necessary corrective actions.

Seabrook has not needed to implement any new technologies to perform inspections of any inaccessible areas at this time. However, Seabrook actively participates in various nuclear utility

owners groups and ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its applicability to Seabrook. Adjustments to inspection plans and availability of new, commercially available technologies for the examination of the inaccessible areas of the containment would be explored and considered as part of these activities.

3.2.6 Containment Coatings Inspections

The site Protective Coatings Program in Specification S-S-1-E-0147 defines the requirements and responsibilities for a program to implement inspections during refueling outages for the purpose of assessing the condition of the protective coatings on structures and equipment in the reactor containment building. These inspections assure compliance with NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors."

Containment coatings inspections are a scheduled activity conducted during refueling outages. The examination areas are selected such that all accessible painted surfaces are inspected every outage.

<u>Results of Recent Coatings Inspections</u> - Fall 2015 Refueling Outage

In general, the overall coatings on steel and concrete substrates did not show any indications of significant degradation and there were two (2) new coated surfaces to be added to the unqualified inventory calculation. The addition does not exceed allowable unqualified coatings inventory. There were no signs of new distress in coatings such as blistering, bubbling, new craze cracking, or discoloration resulting in aging of the coating material. The coatings on concrete walls and floor slabs were in general tightly adhered to the concrete. Cracking does exist in the coating film in several areas where concrete shrinkage and other concrete cracks are present in the concrete. The coating on the edges of the cracks was tightly adhered to the concrete. The coating on the Equipment Hatch and Personnel Hatch and Air Lock were acceptable.

The coating on the liner at all elevations was in excellent condition except for the minor indications in the coating that were reported during the ASME Section XI, IWE examinations. The liner degradation is discussed in Section 3.1.

Conclusion

Based on the observations during the OR17 Containment Coatings Inspection, several items were recommended for remediation or re-coat during OR18 in April 2017.
The results of the coatings inspections concluded the following: (a) the overall condition of the coating systems have not degraded or shown signs of aging since the inspection during OR16 in April 2014, (b) the as left coating conditions are acceptable for continued operation.

3.2.7 License Renewal Commitments

License renewal activities led to two commitments related to the Seabrook containment. The following provides a status of the actions that were identified and committed to by Seabrook in the Seabrook License Renewal Application.

Seabrook Commitment 50:

Perform UT of the accessible areas of the containment liner plate in the vicinity of the moisture barrier for loss of material. Perform opportunistic UT of inaccessible areas.

Status:

Baseline inspections were completed during OR16 (Spring 2014). Repeat containment liner UT thickness examinations at intervals of no more than five (5) refueling outages.

Results:

In the 51 areas the observed minimum thickness ranged from 0.368" to 0.405". The examination areas did not exhibit any signs of corrosion on the opposite side of the steel liner.

Per ASTM A20 the permitted manufacturing tolerance for under thickness conditions of A516 Grade 60 steel plate is 0.010". The allowable reduction in thickness per ASME Section XI, IWE-3122.3(a) is 10% of the nominal plate thickness. The observed minimum thicknesses meet both requirements. In total the examination coverage achieved was greater than 48 square feet of liner plate.

Seabrook Commitment 52:

Implement measures to maintain the exterior surface of the Containment Structure, from elevation -30 feet to +20 feet, in a dewatered state.

Status:

Seabrook verifies at least once per week, either by visual inspection or remotely using the installed camera that the pump is maintaining the area dewatered.

3.3 NRC INFORMATION NOTICE 92-20, "INADEQUATE LOCAL LEAK RATE TESTING"

NRC IN 92-20 was issued to alert licensees to problems involving local leak rate testing of containment penetrations under 10 CFR 50, Appendix J. Problems were identified with the testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

The fuel transfer tube bellows of penetration X-62 are two-ply stainless steel. The LLRT procedure tests flow through the bellows by opening both plugs that are 180 degrees apart from each other, then caps one plug to do the LLRT. This OE was incorporated into the test method at Seabrook following IN 92-20.

3.4 NRC LIMITATIONS AND CONDITIONS

3.4.1 June 25, 2008 NRC Safety Evaluation

The limitations and conditions from the June 25, 2008 safety evaluation to NEI 94-01 Revision 2 are presented in the table below with the NextEra Energy Seabrook response.

June 25, 2008 NRC Safety Evaluation (SE) Limitations and Conditions

Limitation/Condition (From Section 4.0 of Safety Evaluation)	Response for Seabrook
1. For calculating the Type A leakage rate, the licensee should use the definition in NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1).	Seabrook will utilize the definition in NEI 94-01, Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.
2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3)	Reference Section 3.2.1 and 3.2.2. General visual observations of the accessible interior and external surfaces of the containment structure shall continue to be performed in accordance with containment structural integrity test procedures to meet the requirements of the proposed revision to TS 6.15, the inspection requirements of ASME Code Section XI, subsection IWE and NEI 94-01, Revision 3.A, Sections 9.2.1 and 9.2.3.2.
3. The licensee addresses the areas of containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3).	Reference Section 3.2.1 through 3.2.9. General visual observations of the accessible interior and external surfaces of the containment structure shall continue to be performed in accordance with containment structural integrity test procedures to meet the requirements of the proposed revision to TS 6.15, the inspection requirements of ASME Code Section XI, subsection IWE and NEI 94-01, Revision 3.A, Sections 9.2.1 and 9.2.3.2.
4. The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4).	In general, the NRC staff considers the cutting of a large hole in the containment for replacement of steam generators or reactor vessel heads, replacement of large penetrations, as major repair or modifications to the containment structure. Seabrook has performed no major repairs or modifications to the containment structure. No major repairs or modifications are planned.
5. The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provisions of section 9.1 or NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2).	Seabrook will follow the requirements of NEI 94-01, Revision 3-A, Section 9.1. This requirement has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01. In accordance with section 3.1.1.2 of the NRC safety evaluation dated June 25, 2008 (ADAMS Accession No. ML 081140105), NextEra Energy Seabrook will also demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15 year interval is required. Justification for such an extension

Limitation/Condition (From Section 4.0 of Safety Evaluation)	Response for Seabrook
	request will be in accordance with the staff position in Regulatory Issue Summary (RIS) 2008-27.
6. For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design has been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and [Electric Power Research Institute] EPRI Topical Report No. TR-1009325, Revision 2, ["Risk-Impact Assessment of Extended Integrated Leak Rate Testing Intervals,"] including the use of past containment ILRT data.	Not applicable. Seabrook was not licensed under 10 CFR Part 52.

3.4.2 June 8, 2012 NRC Safety Evaluation

The two conditions from Section 4.0 of the June 8, 2012 safety evaluation to NEI 94-01 Revision 3 are stated below with the NextEra Energy Seabrook response.

Condition 1
NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs to be increased to 75 months with the requirement that a licensee's post-outage report include the margin between Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84 months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g., BWR MSIVs), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months. This is Topical Report Condition 1.

Response to Condition 1

Condition 1 presents three (3) separate issues that are addressed as follows:

ISSUE 1 – The allowance of an extended interval for Type C LLRTs of 75 months carries the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit.

Response to Condition 1, Issue 1

The post-outage report shall include the margin between the Type B and Type C minimum pathway leak rate summation value, as adjusted to include the estimate of applicable Type C leakage understatement, and its regulatory limit of 0.60 L_a.

ISSUE 2 – A corrective action plan shall be developed to restore the margin to an acceptable level.

Response to Condition 1, Issue 2

When the potential leakage understatement adjusted Type B and Type C minimum pathway leak rate total is greater than the Seabrook administrative leakage summation limit of 0.50 L_a, but less than the regulatory limit of 0.60 L_a, then an analysis and determination of a corrective action plan shall be prepared to restore the leakage summation margin to less than the Seabrook administrative leakage limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and the manner of timely corrective action (as deemed appropriate) that best focuses on the prevention of future component leakage performance issues.

ISSUE 3 – Use of the allowed 9 month extension for eligible Type C valves is only authorized for non-routine emergent conditions.

Response to Condition 1, Issue 3

Seabrook will apply the 9 month grace period only to eligible Type C components and only for non-routine emergent conditions. Such occurrences will be documented in the record of tests.

<u>Condition 2</u>
The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time. The containment leakage condition monitoring regime involves a portion of the penetrations being tested each refueling outage, nearly all LLRTs being performed during plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leak rates for the just tested penetrations are summed with the as-left minimum pathway leak rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves which, in the aggregate, will show increasing leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate maintenance may extend this increasing leakage potential. Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1. When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations. This is Topical Report Condition 2.

Response to Condition 2

Condition 2 presents two separate issues that are addressed as follows:

ISSUE 1 – Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1.

Response to Condition 2, Issue 1

The change in going from a 60 month extended test interval for Type C tested components to a 75 month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25 percent in the local leak rate test periodicity. As such, NextEra Energy Seabrook will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the as-left leakage total for each Type C component currently on the greater than 60 month (up to 75 month) extended test interval. This will result in a combined conservative Type C total for all 60-75 month local leak rate tests being carried forward and included whenever the total leakage summation is required to be updated (either while operating on-line or following an outage). When the potential leakage understatement adjusted leak rate total for those Type C components being tested on a greater than 60 month (up to 75 month) extended interval is summed with the non-adjusted total of those Type C components being tested at less than the 60-75 month interval and the total of the Type B tested components, if the minimum pathway leak rate is greater than the Seabrook administrative leakage summation limit of 0.50 La, but less than the regulatory limit of 0.60 La, then an analysis and corrective action plan shall be prepared to restore the leakage summation value to less than the administrative leakage limit. The corrective action plan shall focus on those components that have contributed the most to the increase in the leakage summation value and the manner of timely corrective action (as deemed appropriate) that best focuses on the prevention of future component leakage performance issues.

ISSUE 2 – When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Response to Condition 2, Issue 2

If the potential leakage understatement adjusted minimum pathway leak rate is less than the administrative leakage summation limit of 0.50 L_a, then the acceptability of the 75-month local leak rate test extension for all affected Type C components has been adequately demonstrated and the calculated local leak rate total represents the actual leakage potential of the penetrations.

In addition to Condition 1, Issues 1 and 2, which deal with the minimum pathway leak rate Type B and Type C summation margin, NEI 94-01, Revision 3-A, also has the following margin related requirement contained in Section 12.1, "Report Requirements."

A post-outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The technical contents of the report are generally described in ANSI/ANS-56.8-2002 and shall be available on-site for NRC review. The report shall show that the applicable performance criteria are met, and serve as a record that continuing performance is acceptable. The report shall also include the combined Type B and Type C leakage summation, and the margin between the Type B and Type C leakage rate summation and its regulatory limit. Adverse trends in the Type B and Type C leakage rate summation shall be identified in the report and a corrective action plan developed to restore the margin to an acceptable level.

In the event an adverse trend in the potential leakage understatement adjusted Type B and Type C summation is identified, an analysis and a corrective action plan shall be prepared to restore the margin to an acceptable level thereby eliminating the adverse trend. The corrective action plan shall focus on those components that have contributed the most to the adverse trend in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

An adverse trend is defined as three consecutive increases in the Type B and Type C minimum pathway leak rate summation value adjusted to include the estimate of applicable Type C leakage understatement, as expressed in terms of L_a.

3.5 PLANT-SPECIFIC CONFIRMATORY ANALYSIS

3.5.1 Methodology

An evaluation has been performed to assess the risk impact of extending the Seabrook Type A test interval from the current 10 years to 15 years. A simplified bounding analysis consistent with the Electric Power Research Institute (EPRI) approach was used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in:

- Appendix H of Electric Power Research Institute, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325," EPRI Topical Report TR-1018243, dated October 2008,
- Electric Power Research Institute, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," EPRI Topical Report TR-104285, dated August 1994,
- Nuclear Regulatory Commission, "Performance-Based Containment Leak-Test Program," NUREG-1493, dated September 1995, and the
- Calvert Cliffs liner corrosion analysis described in a letter to the NRC dated March 27, 2002 (ADAMS Accession No. ML020920100).

The analysis uses results from Seabrook's Level 1 and Level 2 analysis of core damage scenarios (Level 1) and subsequent containment responses (Level 2) resulting in various fission product release categories (including intact containment or negligible release).

In the safety evaluation issued by NRC letter dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to permanently extend the Type A surveillance test interval to 15 years, subject to the conditions noted in Section 4.2 of the safety evaluation. The following table addresses each of the four conditions for the use of EPRI TR-1009325, Revision 2.

EPRI TR-1009325, Revision 2, Limitations and Conditions

Conditions (From Section 4.2 of Safety Evaluation)	Response for Seabrook
1. The licensee submits documentation indication that the technical adequacy of their (probabilistic risk assessment) PRA is consistent with the requirements of [Regulatory Guide] RG 1.200 relevant to the [integrated leakage rate test] ILRT extension application.	Seabrook PRA technical adequacy is addressed in Section 3.5.2.
2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this [safety evaluation] SE. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in [conditional containment failure probability] CCFP should be defined as a value marginally greater than that accepted in a previous one-time 15 year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points.	EPRI TR-1009325, Revision 2-A, incorporates these population dose and conditional containment failure probability acceptance guidelines, and these guidelines have been used for the Seabrook plant specific assessments. The increase in population dose is discussed in Section 3.5.3. The increase in the conditional containment failure probability is discussed in Section 3.5.3.
3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation in the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak rate accident case (accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.	EPRI TR-1009325, Revision 2-A, incorporates the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident class 3b), and this value has been used in the Seabrook plant specific risk assessment.
4. A [license amendment request] LAR is required in instances where containment over-pressure is relied upon for [emergency core cooling system] ECCS performance.	Seabrook does not rely on containment overpressure for ECCS performance.

3.5.2 Probabilistic Risk Assessment (PRA) Technical Adequacy

The Seabrook PRA model is a fully integrated, full scope Level 1 and Level 2 model, which includes assessment of internal events, internal flood events, internal fire events, and external hazards including seismic events. The Seabrook Station ILRT Risk Impact Assessment uses information from the Seabrook PRA model of record – SSPSS-2014 (Reference 27 of Attachment 4). The Seabrook model of record and supporting documentation has been maintained as a living program, with periodic updates to reflect the as-built, as-operated plant.

The Seabrook PRA model of record SSPSS-2014 meets the requirements of Part 2 "Internal Events" and Part 3 "Internal Flood" of the ASME/ANS PRA Standard and associated NRC Regulatory Guide 1.200 Rev 2. The target capability level for the Seabrook PRA is Capability Category II (CC-II). That is, the goal is to meet all supporting requirements at least at the CC-II level. Note that for many supporting requirements, the requirement spans all three capability categories. Thus, if the SR is met, it meets CC III. While CC II is the target, CC III is met in many SRs.

All significant findings (significance level "A" or "B") from peer reviews, external technical reviews and self-assessments have been addressed and closed. Further, there are no open findings from past peer reviews and self-assessments, for which the PRA did not meet the ASME/ANS PRA Standard Capability Category (CC I) supporting requirements for internal events and internal flood events.

Internal fire events and seismic events have been updated several times and have received several external technical reviews and self-assessments. However, external events have not undergone formal peer review.

The PRA capability review status of Seabrook's PRA models is summarized in Appendix A to Attachment 4.

3.5.3 <u>Conclusion of Plant-Specific Risk Assessment Results</u>

The findings of the Seabrook risk assessment confirm the general findings of previous studies that the risk impact associated with extending the Type A test interval from three in ten years to one in 15 years is small. The Seabrook plant-specific results for extending the Type A test interval from the current 10 years to 15 years is summarized below.

Core damage frequency (CDF) is not impacted by the proposed change. Seabrook does not rely on containment overpressure to assure adequate net positive suction head is available for emergency core cooling system pumps taking suction from the containment sump following design basis accidents.

Regulatory Guide 1.174 (Reference 6.4) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF less than 1E-06 per reactor year and increases in Large Early Release Frequency LERF less than 1E-07 per reactor year.

There is no quantifiable change in CDF as a result of the proposed ILRT Type A test interval extension. Therefore, the delta-CDF is judged to be significantly less than 1E-06 per year and thus meets Regulatory Guide 1.174 acceptance criteria for a "very small" change.

The increase in LERF based on consideration of internal and external events is conservatively estimated at 1.02E-07 per year as a result of the proposed ILRT Type A test 15 year interval.

This result specifically includes consideration of liner corrosion. A delta-LERF of 1.02 E-07 per year is negligibly above (2E-09 per year) the LERF threshold for a "very small" change in risk per Regulatory Guide 1.174 acceptance criteria. Given the conservative, bounding nature of the LERF category 3b evaluation per the EPRI methodology, the incremental change in LERF is judged to pose a "very small" increase in plant risk. In addition, the "total" estimated LERF remains in the range of 2.83E-07 per year, well below the 1E-05 per year total LERF requirement in Regulatory Guide 1.174.

The calculated increase in the total 50-mile population dose risk for the proposed ILRT Type A interval change from three per ten years to once per 15 years is measured as an increase to the total integrated dose risk for all accident sequences. The total 50-mile population dose risk increase (relative to the base case, with corrosion) is 3.82E-02 person-rem per year using the EPRI guidance. It is also noted that the percent change in dose rate between the 3-year interval base case and the 15-year proposed case is estimated at 0.13 percent. EPRI TR-1009325, Revision 2-A, states that a very small population dose is defined as an increase of less than or equal to 1.0 person-rem per year, or less than or equal to one percent of the total population dose, whichever is less restrictive. Thus, the estimated 50-mile population dose increase at Seabrook is very small using the guidelines of EPRI TR-1009325, Revision 2-A.

The increase in the conditional containment failure probability from the three per ten years to once in 15 years Type A test interval including corrosion effects is 0.86 percent for Seabrook. EPRI TR-1009325, Revision 2-A, states that increases in conditional containment failure probability of less than or equal to 1.5 percentage points are very small. Therefore this increase is judged to be very small at Seabrook.

In summary, based on the above results, the proposed 15-year Type A test interval represents a very small change in risk and is acceptable as a permanent change.

Details of the Seabrook risk assessments are contained in Attachment 4 of the LAR.

3.6 CONCLUSION

NEI 94-01, Revision 3-A, describes an NRC accepted approach for implementing the performance-based requirements of Appendix J, Option B. It incorporates the regulatory positions stated in Regulatory Guide 1.163 and includes provisions for extending Type A test intervals to 15 years and Type C test intervals to 75 months. NEI 94-01, Revision 3-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies.

Based on the previous Type A tests conducted at Seabrook, extension of the containment Type A test interval from ten to 15 years represents minimal risk to increased leakage. The risk is further minimized by continued Type B and Type C testing performed in accordance with Appendix J, Option B, and the overlapping inspection activities performed as part of the following Seabrook inspection programs:

- Primary Containment Inservice Inspection Program
- Containment Coatings Inspection and Assessment Program

This experience is supplemented by risk analysis studies, including the Seabrook risk analysis provided in Attachment 4. The findings of the risk assessment confirm the general findings of previous industry studies, and on a plant-specific basis, that extending the Type A test interval

from ten to 15 years results in a very small and acceptable change to the Seabrook Station baseline risk.

4.0 REGULATORY SAFETY ANALYSIS

4.1 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

NextEra Energy Seabrook has evaluated the proposed changes to the Technical Specifications (TS) using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

Description of Amendment Request: An amendment is proposed to the Seabrook Station (SEA) Technical Specification (TS) 6.15, "Containment Leakage Rate Testing Program." The proposed amendment to the TS would revise Seabrook TS 6.15, by replacing the reference to Nuclear Regulatory Commission (NRC) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A, as the implementation document used by Seabrook to implement the performance-based containment leakage rate testing program.

Basis for proposed no significant hazards determination: As required by 10 CFR 50.91(a), the NextEra analysis of the issue of no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," for development of the Seabrook performance-based containment testing program. NEI 94-01 allows, based on risk and performance, an extension of Type A and Type C containment leak test intervals. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less than the values assumed in the plant safety analyses.

The findings of the Seabrook risk assessment confirm the general findings of previous studies that the risk impact with extending the containment leak rate is small. Per the guidance provided in Regulatory Guide 1.174, an extension of the leak test interval in accordance with NEI 94-01, Revision 3-A results in an estimated change within the small change region.

Since the change is implementing a performance-based containment testing program, the proposed amendment does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The requirement for containment leakage rate acceptance will not be changed by this amendment. Therefore,

the containment will continue to perform its design function as a barrier to fission product releases.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change to implement a performance-based containment testing program, associated with integrated leakage rate test frequency, does not change the design or operation of structures, systems, or components of the plant.

The proposed changes would continue to ensure containment integrity and would ensure operation within the bounds of existing accident analyses. There are no accident initiators created or affected by these changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

Margin of safety is related to confidence in the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. The proposed change to implement a performance-based containment testing program, associated with integrated leakage rate test frequency, does not affect plant operations, design functions, or any analysis that verifies the capability of a structure, system, or component of the plant to perform a design function. In addition, this change does not affect safety limits, limiting safety system setpoints, or limiting conditions for operation.

The specific requirements and conditions of the TS Containment Leakage Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained. This ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by implementation of a performance-based containment testing program.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.2 PRECEDENT

This license amendment request is similar to Amendment Nos. 293 and 180 approved for Beaver Valley Unit 1 and Unit 2, respectively, on April 8, 2015 (Reference 6.8).

4.3 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The proposed amendment has been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Nuclear Power Reactors." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values and periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. In addition, Appendix J discusses leakage rate test methodology, frequency of testing, and reporting requirements for each type of test.

Regulatory Guide 1.163, "Performance Based Containment Leak Test Program," (September 1995) provides a method acceptable to the NRC for implementing the performance-based option (Option B) of 10 CFR 50, Appendix J. The regulatory positions stated in Regulatory Guide 1.163 (September 1995) as modified by NRC Safety Evaluations of June 25, 2008 (ADAMS Accession No. ML081140105) and June 8, 2012 (ADAMS Accession No. ML121030286) are incorporated in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

The proposed license amendment would revise Seabrook TS 6.15, "Containment Leakage Rate Testing Program," by changing the wording to indicate that the program shall be in accordance with NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A, instead of Regulatory Guide 1.163, "Performance Based Containment Leak Test Program," and the listed Type A test exception.

The purpose of NEI 94-01 is to assist licensees in the implementation of Option B to 10 CFR Part 50, Appendix J. The NRC staff has reviewed NEI 94-01, Revision 3, and found that this guidance, as modified to include two limitations and conditions, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing.

NextEra has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. Based on the foregoing, the proposed amendment will continue to ensure compliance with 10 CFR 50.54(o), and Option B of 10 CFR Part 50, Appendix J.

4.4 CONCLUSIONS

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in compliance with the Commission's

regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

NextEra has evaluated the proposed amendment for environmental considerations. The review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 Nuclear Energy Institute (NEI) Topical Report, 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 (ML12221A202)

6.2 Letter from S. Bahadur (NRC) to B. Bradley (NEI), "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01, Revision 3, 'Industry Guideline for Implementing Performance-based Option of 10 CFR Part 50, Appendix J,' (TAC No. ME2164)," dated June 8, 2012 (ML121030286)

6.3 Nuclear Energy Institute (NEI) Topical Report, 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008 (ML100620847)

6.4 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011 (ML100910006)

6.5 Electric Power Research Institute, TR-1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated August 2007 (ML072970208)

6.6 NRC Information Notice 2004-09, Corrosion of Steel Containment and Containment Liner, April 27, 2004

6.7 NRC Information Notice 2010-12, Containment Liner Corrosion, June 18, 2010

6.8 NRC letter "Beaver Valley Power Station, Unit Nos. 1 and 2 - Issuance of amendment Re: License Amendment Request to Extend Containment Leakage Rate Test Frequency (TAC Nos. MF3985 and MF 3986)," April 8, 2015 (ML 15078A058)

ATTACHMENT 2

Markup of the Technical Specifications

6.14.1 (Continued)

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A. ~~the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995," as modified by the following exception:~~

~~a. NEI 94-01-1995, Section 9.2.3: The first ILRT performed after October 30, 1992 shall be performed no later than April 29, 2008.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

ATTACHMENT 3

Revised (clean) TS Page

6.14.1 (Continued)

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the SORC.

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A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

ATTACHMENT 4

Risk Impact Assessment

Seabrook Station

Permanent ILRT Interval Extension Risk Impact Assessment

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1 Purpose of Analysis

1.1 Purpose

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) to a permanent fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for Seabrook Station. The risk assessment follows the guidelines from NEI 94-01 (Reference 1), the methodology used in EPRI TR-104285 (Reference 2), the NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 (Reference 3), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 4), the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion induced leakage of steel liners going undetected during the extended test interval (Reference 5), and the methodology used in EPRI 1009325, Revision 2-A (Reference 20).

1.2 Background

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage rate was less than the limiting containment leakage rate of 1La[1].

The basis for the current fifteen year test interval is provided in Section 11.0 of NEI 94-01, Revision 3-A, and was established in 2008. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 (Reference 6), provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR

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[1] La (percent/24 hours) is the maximum allowable leakage rate at pressure Pa (calculated peak containment internal pressure related to the design basis accident) as specified in the technical specifications.

plant (i.e., Surry) containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for Seabrook Station.

The Guidance provided in Appendix H of EPRI Report No. 1009325, "*Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,*" (Reference 20) for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment. The associated change to NEI 94-01 will require that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.3 Criteria

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this permanent extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in Core Damage Frequency (CDF) less than 10^{-6} per reactor year and increases in Large Early Release Frequency (LERF) less than 10^{-7} per reactor year. Therefore, the increase in the conditional containment failure probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the Conditional Containment Failure Probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated. The criteria described below are taken from the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 23).

Regarding CCFP, the NRC concluded that a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one time fifteen year ILRT extension requests. To this end the NRC has endorsed a small increase in CCFP as an increase in CCFP be less than or equal to 1.5% (Reference 23).

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. The NRC concluded that for purposes of assessing the risk impacts of the Type A ILRT extension in accordance with the EPRI methodology, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive (Reference 23).

2 Methodology

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in Appendix H of EPRI Report No. 1009325, Revision 2-A, "*Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*" (Reference 20), EPRI TR-104285 (Reference 2), NUREG-1493 (Reference 6) and the Calvert Cliffs liner corrosion analysis (Reference 5). The bounding analysis uses the current Seabrook Station Level 2 PRA models to establish the baseline containment fission product release categories and associated frequencies.

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report No. 1009325, Revision 2-A (Reference 20).

2. Develop plant specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.

3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.

4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 (Reference 4) and compare with the acceptance guidelines of RG 1.174.

5. Determine the impact of the ILRT interval extension on the Conditional Containment Failure Probability (CCFP) and the population dose and compare with the acceptance guidance of Reference 23.

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore:

- Consistent with the other industry containment leak risk assessments, the Seabrook Station assessment uses LERF and delta LERF in accordance with the risk acceptance guidance of

RG 1.174. Changes in population dose and conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.

- The evaluation for Seabrook Station uses ground rules and methods to calculate changes in risk metrics that are similar to those used in Appendix H of EPRI Report No. 1009325, Revision 2-A, "*Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*."

3 Ground Rules

The following ground rules are used in the analysis:

- The current Seabrook Station Level 1 and Level 2 PRA models are used in this ILRT risk assessment to conservatively estimate the impact that the proposed ILRT extension could have on the Level 2 fission product releases and Large Early Release Frequency (LERF). The Seabrook PRA model is a fully integrated model including internal events, internal flood events, internal fire events, and external hazards including seismic events.

- The technical adequacy of the Seabrook Station PRA model is assessed against RG-1.200, Revision 2, in Appendix A to this report. As discussed in Appendix A, the Seabrook PRA models for internal events and internal flood events meet RG-1.200, Rev 2. The external events models (internal fire events, seismic events and other events) have not been formally peer reviewed against RG-1.200, Rev. 2. Accordingly, the ILRT risk impacts from external events are supplemented with qualitative and sensitivity assessments to further support the conclusion that the potential risk increase associated with the ILRT interval extension is acceptable.

- Dose results for the containment failures modeled in the PRA are obtained from the Severe Accident Mitigation Alternatives Analysis for the NextEra Seabrook Station License Renewal Application (Reference 26).

- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology (Reference 2) and are summarized in Section 4.2.

- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.

- The representative containment leakage for Class 3a sequences is 10La based on the previously approved methodology performed for Indian Point Unit 3 (Reference 8 and Reference 9).

- The representative containment leakage for Class 3b sequences is 100La based on the guidance provided in EPRI Report No. 1009325, Revision 2-A.

- The Class 3b is very conservatively categorized as LERF based on the previously approved methodology (References 8 and 9).

- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

4 Inputs

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 (Reference 10)
2. NUREG/CR-4220 (Reference 11)
3. NUREG-1273 (Reference 12)
4. NUREG/CR-4330 (Reference 13)
5. EPRI TR-105189 (Reference 14)
6. NUREG-1493 (Reference 6)
7. EPRI TR-104285 (Reference 2)
8. NUREG-1150 (Reference 15) and NUREG/CR-4551 (Reference 7)
9. NEI Interim Guidance (Reference 3, Reference 17)
10. Calvert Cliffs Liner Corrosion Analysis (Reference 5)
11. EPRI Report No. 1009325, Revision 2-A, Appendix H (Reference 20)

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50 mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for Seabrook Station. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses

the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent fifteen year extension of the ILRT interval.

4.1.1 NUREG/CR-3539 (Reference 10)

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 (Reference 16) as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

4.1.2 NUREG/CR-4220 (Reference 11)

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage.

4.1.3 NUREG-1273 (Reference 12)

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

4.1.4 NUREG/CR-4330 (Reference 13)

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

"...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

4.1.5 EPRI TR-105189 (Reference 14)

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The conclusion from the study is that a small but measurable safety benefit is realized from extending the test intervals.

4.1.6 NUREG-1493 (Reference 6)

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

Reduction in ILRT frequency from three per ten years to one per twenty years results in an "imperceptible" increase in risk.

Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

4.1.7 EPRI TR-104285 (Reference 2)

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

"... the proposed CLRT (containment leak rate tests) frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.04 person-rem per year ..."

4.1.8 NUREG-1150 (Reference 15) and NUREG/CR 4551 (Reference 7)

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50 mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. However, these

references were not used in this analysis, and Seabrook specific offsite consequences are employed.

4.1.9 NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals (Reference 3, Reference 17)

The guidance provided in this document builds on the EPRI risk impact assessment methodology (Reference 2) and the NRC performance-based containment leakage test program (Reference 6), and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

4.1.10 Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension (Reference 5)

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms were factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder, dome and a concrete basemat, each with a steel liner. Licensees may consider approved LARs for one-time extensions involving containment types similar to their facility. The Seabrook Station assessment has addressed the plant-specific differences from the Calvert Cliffs design, and how the Calvert Cliffs methodology was adapted to address the specific design features.

4.1.11 EPRI Report No. 1009325, Revision 2-A, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (Reference 20)

This report provides a generally applicable assessment of the risk involved in extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant specific supplemental risk impact assessments and builds on the previous EPRI risk impact assessment methodology (Reference 2) and the NRC performance-based containment leakage test program (Reference 6), and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the Seabrook Station assessments to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Classes 3a and 3b scenarios in this analysis as described in Section 5.

4.2 Plant Specific Inputs

The plant specific information used to perform the Seabrook Station ILRT Extension Risk Assessments include the following:

- Level 1 Model results

- Level 2 Model results
- Release category definitions used in the Level 2 Model
- Population within a 50 mile radius
- ILRT results to demonstrate adequacy of the administrative and hardware issues
- Containment failure probability data

4.2.1 Level 1 Model

The Level 1 PRA models that are used for Seabrook Station are characteristic of the as-built plant. The current Seabrook Station model is a linked event tree model, and was quantified for all internal and external initiating events (Internal Events, External Flood, Internal Fire, Internal Flood, Seismic Events, and Severe Weather) with the total Core Damage Frequency (CDF) = 1.20E-05/yr using a truncation value of 1.00E-14. Table 4-1 below summarizes the CDF results (Reference 25).

Table 4-1: Seabrook Station CDF Contribution from External and Internal Event Initiators (Modes 1 to 3)		
Hazard Groups	**CDF for Hazard Group (1/yr)**	**Percent CDF**
Internal Events	3.76E-06	31.33%
External Flood	2.09E-08	0.17%
Internal Fire	1.48E-06	12.33%
Internal Flood	2.86E-06	23.83%
Seismic Events	3.25E-06	27.08%
Severe Weather (LOSP due to severe weather)	6.36E-07	5.30%
CDF Total	1.20E-05	100.00%

4.2.2 Level 2 Model

The Level 2 Model that is used for Seabrook Station was developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 4-2 and Table 4-3 summarize the pertinent Seabrook Station results in terms of release category (Reference 25).

Table 4-2: Seabrook Station Level 2 Release Categories and Frequencies		
Release Category	**Description**	**Frequency/yr[1]**
INTACT1	Containment intact with less than Tech Spec allowed leakage	7.87E-06
INTACT2	Containment intact with greater than Tech Spec allowed leakage	7.73E-08
SE1	Containment bypass via SG tube rupture with feed flow to the faulted SG (scrubbed release)	5.56E-07
SE2	Containment bypass via interfacing LOCA through RHR pump seals (scrubbed release)	2.82E-08
SE3	Containment isolation failure (small penetration) with long term containment cooling	2.30E-07
LL3	Long term controlled containment vent	2.71E-07
LL4	Long term containment overpressure failure	1.03E-06
LL5	Long term containment basemat failure	9.99E-07
SELL	Long term containment failure with initial containment isolation failure (small penetration)	7.50E-07
LE1	Containment bypass via SG tube rupture melt with no feed flow to faulted SG (unscrubbed release)	4.36E-08
LE2	Containment bypass via interfacing LOCA through RHR pipe rupture (unscrubbed release)	1.81E-08
LE3	Containment isolation failure (large penetration, COP valves)	8.59E-10
LE4	Long term containment basemat failure with delayed evacuation	9.20E-08
LERF (LE1 through LE4)	Total LERF release frequency	1.55E-07

Notes:

1. These values were quantified using a truncation value of 1.00E-14.

Table 4-3 summarizes all of the Level 2 release categories and frequencies.

Table 4-3: Seabrook Station Level 2 Release Category Summaries and Frequencies		
Release Category	**Definition**	**Frequency/yr[1]**
INTACT	Containment Intact	7.95E-06
SERF	Small Early Release	8.14E-07
LATE	Late Release	3.05E-06
LERF	Total Large Early Releases	1.55E-07
CDF (including uncategorized releases)[2]		1.20E-05

Notes:

1. These values were quantified using a truncation value of 1.00E-14.
2. This value was calculated as a sum of all release categories (INTACT, SERF, LATE, and LERF).

4.2.3 Population Dose Calculations

The baseline population dose for Seabrook Station is given in Table 4-4 (Reference 25).

Table 4-4: Seabrook Station Release Category Public Dose	
Release Category	**Population Dose (Person-Rem)**
LE1	1.26E+07
LE2	4.27E+07
LE3	2.41E+07
LE4	1.11E+07
SE1	2.43E+05
SE2	8.60E+06
SE3	1.36E+06
LL3	3.63E+06
LL4	7.27E+06
LL5	1.03E+07
SELL	1.48E+07
INTACT1	2.62E+03
INTACT2	1.79E+04

4.2.4 Application of Seabrook Station PRA Model Results to the EPRI Accident Classes

The Seabrook Station Level 2 release categories and frequencies are shown in Table 4-5, along with the corresponding EPRI classes they have been assigned to. The EPRI classes and descriptions are listed in Table 4-6.

Table 4-5: Seabrook Station Level 2 Model Assumptions for Application to the EPRI Accident Classes			
Seabrook Station Level 2 Release Category	**Frequency**	**Definition**	**EPRI Class**
INTACT	7.95E-06	Containment Intact	1
SE1	5.56E-07	Containment bypass via SG tube rupture with feed flow to the faulted SG (scrubbed release)	8
SE2	2.82E-08	Containment bypass via interfacing LOCA through RHR pump seals (scrubbed release)	8
SE3	2.30E-07	Containment isolation failure (small penetration) with long term containment cooling	2
Late	3.05E-06	Long term controlled containment vent	7
LE1	4.36E-08	Containment bypass via SG tube rupture melt with no feed flow to faulted SG (unscrubbed release)	8
LE2	1.81E-08	Containment bypass via interfacing LOCA through RHR pipe rupture (unscrubbed release)	8
LE3	8.59E-10	Containment isolation failure (large penetration, COP valves)	2
LE4	9.20E-08	Long term containment basemat failure with delayed evacuation	7

4.2.5 Release Category Definitions

Table 4-6 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology (Reference 2). These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

Table 4-6: EPRI Containment Failure Classification (Reference 2)	
Class	**Description**
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values La, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.

Table 4-6: EPRI Containment Failure Classification (Reference 2)	
Class	**Description**
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

4.3 Impact of Extension on Detection of Component Failures That Lead to Leakage (Small and Large)

The ILRT can detect a number of component failures such as liner breach, failure of certain bellow arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 4-6, it is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the EPRI Guidance (Reference 20). For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 "small" failures in 217 tests leads to 2/217=0.0092). For Class 3b, Jefferys non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5 / (217+1) = 0.0023).

In a follow on letter (Reference 17) to their ILRT guidance document (Reference 3), NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC Regulatory Guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative

guidance for delta LERF. NEI describes ways to demonstrate that, using plant specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> *The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.*

The application of this additional guidance to the analysis for Seabrook Station, as detailed in Section 5, involves the following:

- The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large preexisting containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Seabrook Station Level 2 PRA analyses.

- Class 1 accident sequences may involve availability and or successful operation of containment sprays. It could be assumed that, for calculation of the Class 3b and 3a frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. However, in this assessment Seabrook Station does not credit containment spray as a means of reducing releases from Class 3 events.

Consistent with the NEI Guidance (Reference 3), the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three year test interval is 1.5 years (3 yr/2), and the average time that a leak could exist without detection for a ten year interval is five years (10 yr/2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. An extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak compared to a three year interval.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC (Reference 9)) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

4.4 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis (Reference 5). The Calvert Cliffs analysis was performed for a concrete cylinder, dome and a concrete basemat, each with a steel liner. Seabrook Station has a similar type of containment.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of a containment steel liner. It should be noted that this computation is being applied to provide an upper bound approach to quantify corrosion induced risk. The Calvert Cliffs corrosion likelihood methodology is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the upper containment (cylinder and dome regions in Calvert Cliffs evaluation)
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

4.4.1 Assumptions

- Consistent with the Calvert Cliffs analysis, a half failure is conservatively assumed for basemat concealed liner corrosion due to the lack of identified failures (See Table 4-7, Step 1). This assumption includes the results of the engineering evaluation (Reference 34) documenting integrity of the containment liner plate as a result of the examination of the moisture barrier in response to the NRC Information Notice IN 2014-07 (Reference 35).

  This report concludes that that the liner plate has maintained its ability to perform its specified leak-tight design function, supported by the following information:

  - The qualitative visual inspections performed did not show any significant degradation of the liner plate.
  - Rate of corrosion for one operating cycle is minimal, 0.0025 in/yr.
  - In 2008, an ILRT was performed on the containment structure and the results of that test showed that the containment structure was performing its specified design function.
  - Examination results indicate that no appreciable corrosion has occurred since the last ILRT in 2008, and there is no containment integrity issue due to corrosion.

- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the Seabrook containment analysis. The events included in the Calvert Cliffs corrosion assessment process, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the nonvisible (backside) portion of the containment liner.

- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is based on 70 steel-lined containments.

- The Calvert Cliffs analysis used the estimated historical liner flaw probability of 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date. Since the time of the Calvert Cliffs submittal, two additional relevant liner corrosion events involving concealed corrosion (corrosion initiated on the inaccessible liner surface) were observed and are considered in the corrosion risk assessment. These events occurred at Beaver Valley Unit 1 and D.C. Cook Unit 2 (Reference 21 and Reference 22, respectively). Consistent with the addition of the two observed events, the historical liner flaw probability was established by incrementing the flaw observation time by 7.75 years. This re-evaluation resulted in a reduction of the historical liner flaw likelihood to 4.3E-03/year ((2+2) / [70 * (5.5 + 7.75)] = 4.3E-03/year). This value is smaller than the value of 5.2E-03 which is used in the Calvert Cliffs analysis. The conservative value of 5.2E-03 will be used in this Seabrook Station report to remain consistent with the Calvert Cliffs analysis.

- Consistent with the Calvert Cliffs analysis, the steel plate flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel ages. (See Table 4-7, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every ten years and every two years.

- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For Seabrook Station, the containment failure probabilities are less than these values at 37 psig based on the containment fragility information which is documented in the Seabrook Station Level 2 analyses. A containment bypass model is utilized for LERF. Conservative probabilities of 1% for an existing through wall flaw to be present on the liner above the basemat and 0.1% for the basemat are used in this analysis, and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude (See Table 4-7, Step 4).

- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the upper containment region (See Table 4-7, Step 4).

- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection (See Table 4-7, Step 5). Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.

- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in large early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

4.4.2 Analysis

<table>
<tr><th colspan="6">Table 4-7: Steel Liner Corrosion Base Case</th></tr>
<tr><th>Step</th><th>Description</th><th colspan="2">Upper Containment</th><th colspan="2">Containment Basemat</th></tr>
<tr><td>1</td><td>Historical Steel Liner Flaw Likelihood

Failure Data: Containment location specific</td><td colspan="2">Events: 2

(2)/(70 * 5.5) = 5.2E-03</td><td colspan="2">Events: 0
(assume half a failure)

0.5/(70 * 5.5) = 1.3E-03</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Age Adjusted Steel Liner Flaw Likelihood

During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).</td><td>Year
1
avg 5-10
15</td><td>Failure Rate
2.1E-03
5.2E-03
1.4E-02</td><td>Year
1
avg 5-10
15</td><td>Failure Rate
5.0E-04
1.3E-03
3.5E-03</td></tr>
<tr><td colspan="2">15 year average = 6.27E-03</td><td colspan="2">15 year average = 1.57E-03</td></tr>
<tr><td>3</td><td>Flaw Likelihood at 3, 10, and 15 years

Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference 5).</td><td colspan="2">0.71% (1 to 3 years)
4.06% (1 to 10 years)
9.40% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the intervals of concern in this analysis.)</td><td colspan="2">0.18% (1 to 3 years)
1.02% (1 to 10 years)
2.35% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the intervals of concern in this analysis.)</td></tr>
</table>

Table 4-7: Steel Liner Corrosion Base Case			
Step	**Description**	**Upper Containment**	**Containment Basemat**
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the leakages for the upper containment and the containment basemat as summarized below for Seabrook Station.

Total Likelihood of Non-Detected Containment Leakage Due To Corrosion for Seabrook Station:

At 3 years: 0.00071% + 0.00018% = 0.00089%

At 10 years: 0.0041% + 0.0010% = 0.0051%

At 15 years: 0.0094% + 0.0024% = 0.012%

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. For example, the three in ten year case is calculated as follows:

- The Seabrook Station CDF associated with accidents that are not independently LERF or could never result in LERF are Level 2 Release Categories INTACT, SERF and LATE. Per Table 4-3, the Seabrook Station CDF associated with accidents that are not independently LERF or could never result in LERF is equal to 7.95E-06/yr + 8.14E-07/yr + 3.05E-06/yr = 1.18E-05/yr.

- Per Table 5-3, the EPRI Class 3b frequency is 2.55E-08/yr.

- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 1.18E-05/yr * 0.00089% = 1.05E-10/yr, where 0.00089% was previously shown above to be the cumulative likelihood of non-detected containment leakage due to corrosion at three years.

- The three in ten year Class 3b frequency including the corrosion-induced concealed flaw issue is then calculated as 2.55E-08/yr + 1.05E-10/yr = 2.56E-08/yr.

5 Results

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H, EPRI-TR-104285 (Reference 2) and previous risk assessment submittals on this subject (References 5, 8, 18, 19) have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5-1 lists these accident classes.

The analysis performed examined Seabrook Station specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown of the severe accidents contributing to risk is considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).

- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test

components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.

- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.

- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-1: Accident Classes

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (Liner Breach)
3b	Large Isolation Failures (Liner Breach)
4	Small Isolation Failures (Failure to Seal-Type B)
5	Small Isolation Failures (Failure to Seal-Type C)
6	Other Isolation Failures (e.g., Dependent Failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including Very Low and No Release)

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5-1.

Step 2 - Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact of extending Type A test interval from three to fifteen and ten to fifteen years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP).

5.1 Step 1 - Quantify the Base-Line Risk in Terms of Frequency Per Reactor Year

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5-1 were developed for Seabrook Station by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 4-5, scaling these frequencies to account for the uncategorized sequences, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion per the methodology described in Section 4.4.

For Seabrook Station, the total frequency of the categorized sequences is 1.20E-05/yr, the total CDF[2] is 1.20E-05/yr, and the scale factor is 1.0. The scaling factor is determined by dividing the total core damage frequency (including the uncategorized frequency) by the total categorized release category frequency (1.20E-05/yr / 1.20E-05/yr = 1.0).

Table 5-2: Seabrook Station Categorized Accident Classes and Frequencies

EPRI Class	Seabrook Station Release Category	Frequency Based on Categorized Results (per yr)	Adjusted Frequency Using Scale Factor of 1.0 (per yr)
1	Intact containment (INTACT1 and INTACT2)	7.95E-06	7.95E-06
2	Containment Isolation failures (LE03 and SE03)	2.31E-07	2.31E-07

[2] CDF as established from the summation of the Level 2 release classes (see Table 4-1).

Table 5-2: Seabrook Station Categorized Accident Classes and Frequencies			
EPRI Class	**Seabrook Station Release Category**	**Frequency Based on Categorized Results (per yr)**	**Adjusted Frequency Using Scale Factor of 1.0 (per yr)**
7	Late Containment Failure (LATE) and Containment Failure (SE02, SE03, LE04)	3.14E-06	3.14E-06
8	Containment Bypass (SE02 and LE02) and SGTR (SE01 and LE01)	6.46E-07	6.46E-07
Total Frequency		1.20E-05	1.20E-05

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially determined from the Containment Intact Level 2 Release Category listed in Table 4-5 minus the EPRI Class 3a and 3b frequency, which are calculated below.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Containment Isolation Failures listed in Table 5-2.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (in excess of design allowable but <10La) or large (>100La).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage

= 0.0092 [see Section 4.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage

= 0.0023 [see Section 4.3]

As described in Section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions).

Class 3a Frequency = 0.0092 * (CDF - (Class 2 + Class 8))

= 0.0092 * (1.20E-05/yr – (2.31E-07/yr + 6.46E-07/yr)) = 1.02E-07/yr

Class 3b Frequency = 0.0023 * (CDF - (Class 2 + Class 8))

$$=0.0023 * (1.20E\text{-}05/yr - (2.31E\text{-}07/yr + 6.46E\text{-}07/yr)) = 2.55E\text{-}08/yr$$

For this analysis, the associated containment leakage for Class 3a is 10La and for Class 3b is 100La. These assignments are consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type C test components occurs. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution, typically resulting in a failure to close smaller containment isolation valves. All other failure modes are bounded by the Class 2 assumptions. Consistent with guidance provided in EPRI Report No. 1009325, Revision 2-A, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from the Severe Accident Phenomena-Induced Failures Release Category from the Seabrook Station Level 2 results shown in Table 4-5.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from the Containment Bypass Release Category from the Seabrook Station Level 2 results shown in Table 4-5.

5.1.1 Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI-TR-104285 the NEI Interim Guidance, and guidance provided in EPRI Report No. 1009325, Revision 2-A. Table 5-3 summarizes these accident frequencies by accident class for Seabrook Station.

Table 5-3: Radionuclide Release Frequencies as a Function of Accident Class (Seabrook Station Base Case)			
Accident Classes (Containment Release Type)	**Description**	**Frequency (per Rx-yr)**	
		EPRI Methodology	**EPRI Methodology Plus Corrosion[1]**
1	No Containment Failure	7.82E-06	7.82E-06
2	Large Isolation Failures (Failure to Close)	2.31E-07	2.31E-07
3a	Small Isolation Failures (liner breach)	1.02E-07	1.02E-07
3b	Large Isolation Failures (liner breach)	2.55E-08	2.56E-08
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.14E-06	3.14E-06
8	Bypass (Interfacing System LOCA)	6.46E-07	6.46E-07
CDF	All CET end states	1.20E-05	1.20E-05

1. Note that this is based on data developed in Section 4.4. Only Classes 1 and 3b are impacted by the corrosion.

5.2 Step 2 - Develop Plant Specific Person-Rem Dose (Population Dose) Per Reactor Year

Plant specific release analyses were performed to estimate the person-rem doses to the population within a 50 mile radius from the plant, and summarized in Table 4-4. The results of applying these releases to the EPRI containment failure classification are as follows:

Class 1 = 2.77E+03 person-rem (Note 1)

Class 2 = 1.44E+06 person rem (Note 2)

Class 3a = 2.77E+03 person-rem x 10La = 2.77E+04 person-rem (Note 3)

Class 3b = 2.77E+03 person-rem x 100La = 2.77E+05 person-rem (Note 3)

Class 4 = Not analyzed

Class 5 = Not analyzed

Class 6 = Not analyzed

Class 7 = 9.25E+06 person rem (Note 4)

Class 8 = 6.46E+05 person-rem (Note 5)

Notes:

(1) Class 1 is assigned the dose from the frequency weighted average of the dose from release categories Intact1 and Intact2 from Table 4-2 and Table 4-4.

(2) Class 2 is assigned the dose from the frequency weighted average of the dose from release categories LE3 and SE3 from Table 4-2 and Table 4-4.

(3) The Class 3a and 3b dose are related to the Class 1 leakage rate as shown. While no pre-existing leakage in excess of 21 La has been identified for any historical ILRT event, Class 3b releases are conservatively assessed at 100La. Class 3a releases are conservatively assessed at 10La. This is consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A.

(4) Class 7 is assigned the frequency weighted average of the dose from release categories LE4, LL3, LL4, LL5, and SELL from Table 4-4.

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from the frequency weighted average of the dose from release categories SE1, SE2, LE1 and LE2 from Table 4-4.

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology (Reference 2) containment failure classifications, and consistent with the NEI guidance (Reference 3) as modified by EPRI Report No. 1009325, Revision 2-A are provided in Table 5-4.

Table 5-4: Seabrook Station Population Dose Estimates for Population Within 50 Miles

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	2.77E+03
2	Large Isolation Failures (Failure to Close)	1.44E+06
3a	Small Isolation Failures (liner breach)	2.77E+04
3b	Large Isolation Failures (liner breach)	2.77E+05
4	Small Isolation Failures (Failure to seal-Type B)	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	9.25E+06
8	Bypass (Interfacing System LOCA and Unisolated SGTR)	6.46E+05

The above dose estimates, when combined with the results presented in Table 5-3, yield the Seabrook Station baseline mean consequence measures for each accident class. These results are presented in Table 5-5.

Table 5-5: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr[1]**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person- Rem/yr (50 miles)**	
1	No Containment Failure [2]	2.77E+03	7.82E-06	2.16E-02	7.82E-06	2.16E-02	-2.91E-07
2	Large Isolation Failures (Failure to Close)	1.44E+06	2.31E-07	3.34E-01	2.31E-07	3.34E-01	0.00E+00
3a	Small Isolation Failures (liner breach)	2.77E+04	1.02E-07	2.82E-03	1.02E-07	2.82E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	2.77E+05	2.55E-08	7.06E-03	2.56E-08	7.09E-03	2.91E-05
4	Small Isolation Failures (Failure to seal -Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	9.25E+06	3.14E-06	2.91E+01	3.14E-06	2.91E+01	0.00E+00

Table 5-5: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 3/10 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr$^{(1)}$**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person- Rem/yr (50 miles)**	
8	Bypass (Interfacing System LOCA)	6.46E+05	6.46E-07	4.18E-01	6.46E-07	4.18E-01	0.00E+00
CDF	All CET end states	N/A	1.20E-05	2.99E+01	1.20E-05	2.99E+01	2.88E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.3 Step 3 - Evaluate Risk Impact of Extending Type A Test Interval From 10 to 15 Years

The next step is to evaluate the risk impact of extending the test interval from its current ten year value to fifteen years. To do this, an evaluation must first be made of the risk associated with the ten year interval since the base case applies to a three year interval (i.e., a simplified representation of a three in ten interval).

5.3.1 Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is directly impacted. As it is assumed that the new Class 3 endstates arise from previously intact containment states, the intact state frequency is reduced accordingly. The risk contribution is changed based on the NEI guidance as described in Section 4.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a ten year interval are presented in Table 5-6.

5.3.2 Risk Impact Due to 15-Year Test Interval

The risk contribution for a fifteen year interval is calculated in a manner similar to the ten year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the three year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5-7.

Table 5-6: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years							
Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.77E+03	7.52E-06	2.08E-02	7.52E-06	2.08E-02	-9.69E-07
2	Large Isolation Failures (Failure to Close)	1.44E+06	2.31E-07	3.34E-01	2.31E-07	3.34E-01	0.00E+00
3a	Small Isolation Failures (liner breach)	2.77E+04	3.40E-07	9.41E-03	3.40E-07	9.41E-03	0.00E+00
3b	Large Isolation Failures (liner breach)	2.77E+05	8.49E-08	2.35E-02	8.53E-08	2.36E-02	9.69E-05
4	Small Isolation Failures(Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	9.25E+06	3.14E-06	2.91E+01	3.14E-06	2.91E+01	0.00E+00

Table 5-6: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/10 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr[(1)]**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person-Rem/yr (50 miles)**	
8	Bypass (Interfacing System LOCA)	6.46E+05	6.46E-07	4.18E-01	6.46E-07	4.18E-01	0.00E+00
CDF	All CET end states	N/A	1.20E-05	2.99E+01	1.20E-05	2.99E+01	9.59E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-7: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years							
Accident Classes (Cnmt Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.77E+03	7.31E-06	2.02E-02	7.31E-06	2.02E-02	-1.46E-06
2	Large Isolation Failures (Failure to Close)	1.44E+06	2.31E-07	3.34E-01	2.31E-07	3.34E-01	0.00E+00
3a	Small Isolation Failures (liner breach)	2.77E+04	5.10E-07	1.41E-02	5.10E-07	1.41E-02	0.00E+00
3b	Large Isolation Failures (liner breach)	2.77E+05	1.28E-07	3.53E-02	1.28E-07	3.55E-02	1.46E-04
4	Small Isolation Failures (Failure to seal-Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal-Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

Table 5-7: Seabrook Station Annual Dose as a Function of Accident Class; Characteristic of Conditions for ILRT Required 1/15 Years							
Accident Classes (Cnmt Release Type)	**Description**	**Person-Rem (50 miles)**	**EPRI Methodology**		**EPRI Methodology Plus Corrosion**		**Change Due to Corrosion Person-Rem/yr(1)**
			Frequency (per Rx-yr)	**Person-Rem/yr (50 miles)**	**Frequency (per Rx-yr)**	**Person-Rem/yr (50 miles)**	
7	Failures Induced by Phenomena (Early and Late)	9.25E+06	3.14E-06	2.91E+01	3.14E-06	2.91E+01	0.00E+00
8	Bypass (Interfacing System LOCA)	6.46E+05	6.46E-07	4.18E-01	6.46E-07	4.18E-01	0.00E+00
CDF	All CET end states	N/A	1.20E-05	2.99E+01	1.20E-05	2.99E+01	1.44E-04

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as 1La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 Step 4 - Determine the Change in Risk in Terms of Large Early Release Frequency (LERF)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the EPRI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Seabrook Station, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on a ten year test interval from Table 5-6, the Class 3b large early release frequency contribution (conservatively including corrosion) is 8.53E-08/yr; and, based on a fifteen year test interval from Table 5-7, this LERF contribution increases to 1.28E-07/yr. Thus, the increase in the overall LERF due to Class 3b sequences that is due to increasing the ILRT test interval from three to fifteen years is 1.02E-07/yr as shown in Table 5-8. Similarly, the increase in LERF due to increasing the ILRT interval from ten years to fifteen years is 4.28E-08/yr as shown in Table 5-8.

As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF for Seabrook Station is below the threshold criteria for a very small change when comparing both the fifteen year results to the current ten year requirement, and when the fifteen year ILRT extension results are compared to the original three year requirement, the increase in LERF is negligibly above (2×10^{-9}/year) the threshold for very small change and well below the threshold criteria for small change. See Table 5-8 for more information. In light of the conservative 3b assignment of large liner leaks to LERF, the incremental increase is judged to pose a very small increase to plant risk.

5.5 Step 5 - Determine the Impact on the Conditional Containment Failure Probability (CCFP)

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the EPRI Report No. 1009325, Revision 2-A. The NRC has previously accepted similar calculations (Reference 9) as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. The list below shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects.

CCFP = [1 - (Class 1 frequency + Class 3a frequency) / CDF] * 100%

$CCFP_3$ = [1 - (7.82E-06/yr + 1.02E-07/yr) / 1.20E-05/yr] * 100% = 33.80%

$CCFP_3$ = 33.80%

$CCFP_{10}$ = [1 - (7.52E-06/yr + 3.40E-07/yr) / 1.20E-05/yr] * 100% = 34.30%

$CCFP_{10}$ = 34.30%

$CCFP_{15}$ = [1 - (7.31E-06/yr + 5.10E-07/yr) / 1.20E-05/yr] * 100% = 34.65%

$CCFP_{15}$ = 34.65%

$\Delta CCFP = CCFP_{15} - CCFP_3 = 0.86\%$

$\Delta CCFP = CCFP_{15} - CCFP_{10} = 0.36\%$

$\Delta CCFP = CCFP_{10} - CCFP_3 = 0.50\%$

The change in CCFP of approximately 0.86% by extending the test interval to fifteen years from the original three in ten year requirement is judged to be very small.

5.6 Summary of Results

The results from this ILRT extension risk assessment for Seabrook Station are summarized in Table 5-8.

Table 5-8: Seabrook Station ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.77E+03	7.82E-06	2.16E-02	7.52E-06	2.08E-02	7.31E-06	2.02E-02
2	1.44E+06	2.31E-07	3.34E-01	2.31E-07	3.34E-01	2.31E-07	3.34E-01
3a	2.77E+04	1.02E-07	2.82E-03	3.40E-07	9.41E-03	5.10E-07	1.41E-02
3b	2.77E+05	2.56E-08	7.09E-03	8.53E-08	2.36E-02	1.28E-07	3.55E-02
7	9.25E+06	3.14E-06	2.91E+01	3.14E-06	2.91E+01	3.14E-06	2.91E+01
8	6.46E+05	6.46E-07	4.18E-01	6.46E-07	4.18E-01	6.46E-07	4.18E-01
Total	N/A	1.20E-05	2.99E+01	1.20E-05	2.99E+01	1.20E-05	2.99E+01
ILRT Dose Rate from 3a and 3b Per-Rem/Yr		9.92E-03		3.30E-02		4.96E-02	
Delta Total Dose Rate[1]	From 3 yr	N/A		2.23E-02		3.82E-02	
	From 10 yr	N/A		N/A		1.60E-02	
% change in dose rate from base	From 3 yr	N/A		0.07%		0.13%	
	From 10 yr	N/A		N/A		0.05%	
3b Frequency (LERF) /Yr		2.56E-08		8.53E-08		1.28E-07	
Delta LERF	From 3 yr	N/A		5.97E-08		1.02E-07	
	From 10 yr	N/A		N/A		4.28E-08	
CCFP %		33.80%		34.30%		34.65%	
Delta CCFP %	From 3 yr	N/A		0.50%		0.86%	
	From 10 yr	N/A		N/A		0.36%	

[1] The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

6 Sensitivities

6.1 Sensitivity to Corrosion Impact Assumptions

The Seabrook Station results in Table 5-5 through Table 5-7 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the upper containment and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6-1. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 1.05E-11/yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6-1: Steel Plate Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	**Containment Breach (Step 4 in the corrosion analysis)**	**Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)**	**Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 years (per Rx-yr)**	
			Total Increase	**Increase Due to Corrosion**
Base Case Doubles every 5 yrs	Base Case (1% Upper Containment, 0.1% Basemat)	Base Case (10% Upper Containment, 100% Basemat)	1.02E-07	4.20E-10
Doubles every 2 yrs	Base	Base	1.03E-07	7.49E-10
Doubles every 10 yrs	Base	Base	1.02E-07	1.21E-10
Base	Base	*15%*	1.03E-07	5.88E-10
Base	Base	*5%*	1.02E-07	2.53E-10

Table 6-1: Steel Plate Corrosion Sensitivity Cases				
Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 years (per Rx-yr)	
			Total Increase	Increase Due to Corrosion
Base	10% Upper Containment, 1% Basemat	Base	1.06E-07	4.20E-09
Base	0.1% Upper Containment, 0.01% Basemat	Base	1.02E-07	4.20E-11
Lower Bound				
Doubles every 10 yrs	0.1% Upper Containment, 0.01% Basemat	5% Upper Containment, 1% Basemat	1.02E-07	7.28E-15
Upper Bound				
Doubles every 2 yrs	10% Upper Containment, 1% Basemat	15% Upper Containment, 100% Basemat	1.02E-07	1.05E-11

6.2 Sensitivity to Class 3B Contribution to LERF

The Class 3b frequency for the base case of a three in ten year ILRT interval including corrosion is 2.56E-08/yr (Table 5-5). Extending the interval to one in ten years results in a frequency of 8.53E-08/yr (Table 5-6). Extending it to one in fifteen years results in a frequency of 1.28E-07/yr (Table 5-7), which is an increase of 1.02E-07/yr from three in ten years to once in fifteen years.

If 100% of the Class 3b sequences are assumed to have potential releases large enough for LERF, then the increase in LERF due to extending the interval from three in ten to one in fifteen is slightly above the RG 1.174 threshold for very small changes in LERF of 1.00E-07/yr.

6.3 Potential Impact from External Events Contribution

As described in Section 4.2.1 above, the ILRT risk assessment quantitative results are based on Seabrook Station's fully integrated, full scope Level 1 and Level 2 PRA model. This model includes assessment of internal events, internal flood events, internal fire events, and applicable external hazards including seismic events.

Based on the Seabrook Station PRA Capability, the internal fire events and external events PRA models are detailed, quantitative models judged to provide realistic/best estimate plant risk results and insights. However, these models have not been formally peer reviewed against the requirements of the PRA Standard and RG-1.200, Rev. 2. Accordingly, internal fire and external events risks are qualitatively assessed below to further demonstrate that the proposed ILRT Type A test extension has a minimal risk impact from external events.

Internal Fire Events - Qualitative Risk Impacts on ILRT Extension

The Seabrook Fire PRA includes an integrated quantitative assessment of at-power internal fire risk. In the base case plant risk model, internal fire events contribute approximately 12% to the total at-power CDF. The base case quantitative CDF/LERF results for internal fire events are as follows:

Fire CDF: 1.48E-06/yr

Fire LERF: 1.27E-10/yr

The proposed extension of the ILRT interval does not impact the frequency of any fire initiating events nor does the ILRT extension impact the reliability of active equipment credited in fire initiating event sequences. The Seabrook plant is a relatively late vintage design with good separation design between divisional equipment. This limits the number of important fire areas in the plant. Important fire sequences include fire events in the control room (CR) and cable spreading room (CSR) (which require abandonment and remote safe-shutdown actions) and fire events in the essential switchgear rooms (ESWGR). Dominant fire initiators include the following:

CR or CSR fire causing PORV to open

ESWGR fire (in either room) causing a loss of bus E5 or E6

CR or CSR fire with loss of AC power

Fire events generally result in postulated fire damage and failure of one equipment division. These events lead to loss of support system functions through additional random failures resulting in loss of RCP seal cooling and a challenge to RCP seal integrity. Small LOCA-type sequences could also occur during a fire event through postulated spurious opening of a PORV.

The ILRT test is focused on performing a periodic validation of the containment liner leak tightness, which is a condition that is independent of fire events risk. Therefore the ILRT extension has no direct effect on the core damage from fire events.

As shown above, fire events do not contribute significantly to LERF (1.3E-10/yr). This is consistent with Seabrook's relatively low total LERF contribution (1.5E-07/yr) from all internal and external events, with the total LERF of all events being approximately two orders of magnitude lower than total CDF (1.2E-05/yr). The relatively low total LERF is a reflection of Seabrook's robust containment design and associated release mitigation capability. Because Seabrook's baseline LERF is relatively low, the "added" 3b-LERF introduced by the ILRT extension, although very small, is relatively large compared to the low baseline LERF. Thus, the delta-LERF as a result of applying the industry ILRT methodology appears as a relatively large increase for Seabrook.

As noted in Appendix A the Seabrook fire PRA is in the process of being formally upgraded to meet the latest industry methods (guidance in NUREG/CR-6850 (Reference 33) and CC II of the ASME PRA Standard. Insights gained thus far in the update process continue to suggest that the current fire PRA framework is comprehensive and a realistic representation of fire risk. However, as a result of applying the latest industry fire risk methods, it is recognized that the final updated fire-induced CDF/LERF could increase above the current values. Therefore, a fire/ILRT risk sensitivity evaluation was performed by conservatively assuming that the current fire-induced CDF is increased by a factor of 2. This sensitivity updates the calculation of core damage frequency in the analysis shown in Table 4-1, and increases the fire contribution to the release category frequencies in Table 4-2 by a factor of 2. All other calculations in the assessment are unchanged. The results of this evaluation are shown in Table 6-2. This evaluation shows that the 15 year ILRT test interval 3b LERF would become 1.45E-07/yr resulting in a delta-LERF of 1.16E-07/yr. This represents an increase in the delta-LERF of approximately 1.4E-08/yr above the baseline delta-LERF of 1.02E-07/yr. This shows that the baseline delta-LERF is relatively insensitive to the assumed increase in fire risk.

Conclusion - The baseline quantitative fire CDF and LERF contributions are judged appropriate for use in the delta LERF impact of the ILRT Type A test interval extension. No additional fire risk insights were identified for further consideration.

Table 6-2: Senstivity to Doubling of Fire CDF for Seabrook Station

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.77E+03	8.97E-06	2.48E-02	8.64E-06	2.39E-02	8.39E-06	2.32E-02
2	1.45E+06	2.36E-07	3.44E-01	2.36E-07	3.44E-01	2.36E-07	3.44E-01
3a	2.77E+04	1.16E-07	3.20E-03	3.85E-07	1.07E-02	5.78E-07	1.60E-02
3b	2.77E+05	2.90E-08	8.03E-03	9.66E-08	2.68E-02	1.45E-07	4.02E-02
7	9.16E+06	3.44E-06	3.15E+01	3.44E-06	3.15E+01	3.44E-06	3.15E+01

Table 6-2: Senstivity to Doubling of Fire CDF for Seabrook Station							
8	6.46E+05	6.46E-07	4.18E-01	6.46E-07	4.18E-01	6.46E-07	4.18E-01
Total	N/A	1.34E-05	3.23E+01	1.34E-05	3.24E+01	1.34E-05	3.24E+01
ILRT Dose Rate from 3a and 3b		1.12E-02		3.74E-02		5.62E-02	
Delta Total Dose Rate	From 3 yr	N/A		2.52E-02		4.33E-02	
	From 10 yr	N/A		N/A		1.81E-02	
% change in dose rate from base	From 3 yr	N/A		0.08%		0.13%	
	From 10 yr	N/A		N/A		0.06%	
3b Frequency (LERF)		2.90E-08		9.66E-08		1.45E-07	
Delta LERF	From 3 yr	N/A		6.76E-08		1.16E-07	
	From 10 yr	N/A		N/A		4.85E-08	
CCFP %		32.40%		32.90%		33.26%	
Delta CCFP%	From 3 yr	N/A		0.50%		0.86%	
	From 10 yr	N/A		N/A		0.36%	

Seismic Events - Qualitative Risk Impacts on ILRT Extension

The Seabrook Seismic hazards PRA includes a quantitative assessment of at-power seismic risk. In the base case plant risk model, seismic events contribute approximately 27% to the total at-power CDF. The seismic hazards PRA is based on the probabilistic hazard curves in Reference 30. The base case quantitative CDF/LERF results for seismic events are as follows:

CDF: 3.25E-06/yr

LERF: 9.85E-08/yr

The proposed extension of the ILRT interval does not impact the frequency of any seismic initiating event nor does the ILRT extension impact the reliability of active equipment credited in seismic initiating event sequences. The Seabrook plant is a relatively late vintage design and has a robust structural seismic capacity. Seismic risk is dominated by large, beyond-design-basis seismic events that result in transients with loss of offsite power with failure of the Emergency Diesel Generators and Supplemental Emergency Power Supply diesels (station blackout condition) and failure of other non-seismically supported and seismically supported equipment needed for core cooling. Large seismic events can also result in large LOCA events with seismic-induced failure of ECCS equipment. Smaller size seismic events are less likely to cause a large LOCA or blackout-type event and support systems and ECCS are more likely to remain

available. These lower level seismic event sequences (transient sequences where the equipment survives the seismic event but is then subject to random failure) are important sequences but are less dominant than the large seismic event sequences.

The ILRT test is focused on performing a periodic validation of the containment liner leak tightness, which is a condition independent of the risk of seismic events. Therefore the ILRT extension has no direct effect on the core damage/release mitigation capability from seismic events.

As shown above, seismic events contribute approximately 9.8E-08/yr to LERF. The majority of this contribution is from release category LE4, which conservatively captures large late releases and assigns them as early releases due to the possibility of delayed evacuation. Seabrook's total LERF contribution from all internal and external events is relatively low at 1.55E-07/yr. This is approximately two orders of magnitude lower than total CDF of 1.2E-05/yr. The relatively low total LERF is a reflection of Seabrook's robust containment design. Because Seabrook's baseline LERF is relatively low, the "added"3b-LERF introduced by the ILRT extension, although very small, is relatively large compared to the low baseline LERF. Thus, the delta-LERF as a result of applying the industry ILRT methodology appears as a relatively large increase for Seabrook.

As noted in Appendix A, the Seabrook seismic PRA is judged to provide a realistic representation of the seismic hazard risk from operation of Seabrook Station. However, the SPRA has not been formally peer reviewed against the requirements of the PRA Standard and RG-1.200, Rev. 2. As a result of applying the latest industry seismic risk hazard and methods, it is recognized that the actual Seabrook seismic-induced CDF/LERF could increase above the current values. Therefore, a seismic/ILRT risk sensitivity evaluation was performed by assuming that the current seismic-induced CDF is equivalent to 1E-05/yr, taken from Table D-1 of Appendix D to, "Results of Safety/Risk Assessment of Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants (Reference 29). This sensitivity updates the calculation of core damage frequency in the analysis shown in Table 4-1, and increases the seismic contribution to the release category frequencies in Table 4-2 by a factor of 3.08 (3.25E-06/yr to 1.0E-05/yr). All other calculations in the assessment are unchanged. The results of this evaluation are shown in Table 6-3. This evaluation shows that the 15 year ILRT test interval 3b LERF would become 2.01E-07/yr resulting in a delta-LERF of 1.61E-07/yr. This represents an increase in the delta-LERF of approximately 6E-08/yr above the baseline delta-LERF of 1.02E-07/yr. This shows that the baseline delta-LERF is relatively insensitive to the assumed increase in seismic risk.

Conclusion - The baseline quantitative seismic CDF and LERF contributions are judged appropriate for use in the delta LERF impact of the ILRT Type A test interval extension. No additional seismic risk insights are identified for further consideration.

Table 6-3: Senstivity to GI-199 Seismic CDF for Seabrook Station

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.77E+03	1.08E-05	3.00E-02	1.04E-05	2.87E-02	1.00E-05	2.78E-02
2	1.41E+06	5.81E-07	8.17E-01	5.81E-07	8.17E-01	5.81E-07	8.17E-01
3a	2.77E+04	1.60E-07	4.44E-03	5.34E-07	1.48E-02	8.02E-07	2.22E-02
3b	2.77E+05	4.02E-08	1.11E-02	1.34E-07	3.71E-02	2.01E-07	5.57E-02
7	9.77E+06	6.40E-06	6.25E+01	6.40E-06	6.25E+01	6.40E-06	6.25E+01
8	6.41E+05	6.68E-07	4.28E-01	6.68E-07	4.28E-01	6.68E-07	4.28E-01
Total	N/A	1.87E-05	6.38E+01	1.87E-05	6.38E+01	1.87E-05	6.38E+01
ILRT Dose Rate from 3a and 3b		1.56E-02		5.19E-02		7.79E-02	
Delta Total Dose Rate	From 3 yr	N/A		3.50E-02		6.01E-02	
	From 10 yr	N/A		N/A		2.51E-02	
% change in dose rate from base	From 3 yr	N/A		0.05%		0.09%	
	From 10 yr	N/A		N/A		0.04%	
3b Frequency (LERF)		4.02E-08		1.34E-07		2.01E-07	
Delta LERF	From 3 yr	N/A		9.38E-08		1.61E-07	
	From 10 yr	N/A		N/A		6.72E-08	
CCFP %		41.15%		41.66%		42.02%	
Delta CCFP%	From 3 yr	N/A		0.50%		0.86%	
	From 10 yr	N/A		N/A		0.36%	

Other External Events - Qualitative Risk Impacts on ILRT Extension

Other external events include LOSP due to severe weather, external flooding, high winds, tornado, transportation and near-by facility hazards, turbine missile, etc. Some of these events (e.g., weather-related LOSP) are quantified in the internal events PRA model while others are screened out based on the low probability of occurrence and rigorous plant design features. Collectively, all "other" external events are judged to have a very small contribution to CDF/LERF. The proposed extension of the ILRT test interval does not impact the initiating event frequencies of other external events nor does the ILRT extension impact the reliability of active equipment needed to mitigate these events. The ILRT test is focused on performing a periodic validation of the containment liner leak tightness, which is a condition that is independent of external event risk. Therefore the ILRT extension has no direct effect on the core damage and release mitigation capability from external events.

Conclusion – Other external events are judged to have a very small contribution to CDF/LERF and are screened from detailed quantitative analysis. As a result, these events are judged to have negligible risk impact on the proposed ILRT Type A test interval extension. No additional other external events risk insights are identified for further consideration.

7 Conclusions

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the following conclusions regarding the assessment of the plant risk are associated with permanently extending the Type A ILRT test frequency to once in fifteen years:

- Regulatory Guide 1.174 (Reference 4) provides guidance for determining the risk impact of plant specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is conservatively estimated as 1.02E-07/yr which is marginally above the RG 1.174 "very small" risk criterion. In light of the conservative assignment of 3b large liner leaks to LERF the incremental increase is judged to pose a very small increase to plant risk using the acceptance guidelines of Reg. Guide 1.174.

- Regulatory Guide 1.174 (Reference 4) also states that when the calculated increase in LERF is in the range of 1.00E-07 per reactor year to 1.00E-06 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.00E-05 per reactor year. The assessment includes the impact from External Events. In this case, the total class 3b contribution to LERF including External Events was conservatively estimated as 1.28E-07/yr for Seabrook Station. The resulting total LERF is 1.28E-07/yr + 1.55E-07/yr (from Table 4-2) = 2.83E-07/yr. This is below the RG 1.174 acceptance criteria for total LERF of 1.00E-05/yr and therefore this change satisfied both the incremental and absolute expectations with regard to the RG 1.174 LERF metric.

- The change in Type A test frequency to once per fifteen years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 3.82E-02 person-rem/yr. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of ≤1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 23). Moreover, the risk impact when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure probability from the three in ten year interval to a permanent one time in fifteen year interval is 0.86%. EPRI Report

No. 1009325, Revision 2-A states that increases in CCFP of ≤1.5 percentage points are very small. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325 (Reference 23). Therefore this increase is judged to be very small.

Therefore, permanently increasing the ILRT interval to fifteen years is considered to be a very small change to the Seabrook Station risk profile.

7.1.1 Previous Assessments

The NRC in NUREG-1493 (Reference 6) has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per ten years to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage rate tests is possible with minimal impact on public risk. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for Seabrook Station confirm these general findings on a plant specific basis considering the severe accidents evaluated for Seabrook Station, the Seabrook Station containment failure modes, and the local population surrounding Seabrook Station.

8 References

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Appendix A

Seabrook Station Permanent ILRT Interval Extension Risk Impact Assessment: Seabrook Station PRA Capability

Introduction

The Seabrook PRA model is a fully integrated, full scope Level 1 and Level 2 model, which includes assessment of internal events, internal flood events, internal fire events, and external hazards including seismic events. The Seabrook Station ILRT Risk Impact Assessment uses information from the Seabrook PRA model of record – SSPSS-2014 (Reference 27). This analysis of Seabrook Station PRA capability is based on information contained in Reference 31.

The Seabrook PRA model of record SSPSS-2014 meets the requirements of Part 2 "Internal Events" and Part 3 "Internal Flood" of the ASME/ANS PRA Standard and associated NRC Regulatory Guide 1.200 Rev 2 (Reference 32). The target capability level for the Seabrook PRA is Capability Category II (CC-II). That is, the goal is to meet all supporting requirements at least at the CC-II level. Note that for many supporting requirements, the requirement spans all three capability categories. Thus, if the SR is met, it meets CC III. While CC II is the target, CC III is met in many SRs.

All significant findings (significance level "A" or "B") from peer reviews, external technical reviews and self-assessments have been addressed and closed. Further, there are no open findings from past peer reviews and self-assessments, for which the PRA did not meet the ASME/ANS PRA Standard Capability Category (CC I) supporting requirements for internal events and internal flood events.

Internal fire events and seismic events have been updated several times and have received several external technical reviews and self assessments. However, external events have not undergone formal peer review.

The PRA capability review status of Seabrook's PRA models is summarized below.

Internal Events (ASME PRA Standard, Part 2)

The internal events portion of the Seabrook PRA has been updated a number of times since the original SSPSA-1983, including the most recent SSPSS-2014 Update.

Three peer reviews have been conducted against internal event supporting requirements:

- In 1999, a review of all technical elements was performed using the industry PSA Certification process, the precursor to the PRA Standard. The 1999 peer review resulted in 2 "A" level and 28 "B" level findings. All of these A and B findings have been addressed and closed.

- In 2005, a focused peer review was performed for the elements AS, SC and HR as well as configuration control. This assessment replaced the 1999 peer

review for those elements that were in scope. This review was done using the then current PRA Standard (ASME RA-Sa-2003). The 2005 focused peer review resulted in 0 "A" findings and 4 "B" findings. All of these A and B findings have been addressed and closed.

- In 2012, a focused peer review was performed for the element LE. This assessment replaced the 1999 peer review for that element. This review was done using the current PRA Standard (ASME/ANS RA-Sa-2009). The 2012 peer review resulted in 6 findings ("F") associated with 10 SRs. All of these findings have been addressed in the 2014 PRA and are closed.

Three external reviews were conducted in 2002, 2006, and 2011 of specific internal events model issues. While these reviews were not specifically against PRA Standard requirements, they represented outside technical reviews which identified observations for improvement, but none were classified at the A or B level.

Five self-assessments against the internal event SRs in the PRA standard have been performed in 2005 (ASME RA-Sa-2003), 2007 (ASME RA-Sb-2005), 2010 (ASME/ANS RA-Sa-2009), 2011 (ASME/ANS RA-Sa-2009) and 2014 (ASME/ANS RA-Sa-2009). The first three self-assessments considered all internal events technical elements. SA-2011 addressed only the open items against specific SRs and SA-2014 addressed all of the SRs for element LE.

Conclusion - All Internal Events peer review significant A and B Findings have been addressed and closed. The 2014 PRA meets all Part 2 (internal events) CC II (or higher) requirements of the PRA Standard, as endorsed by RG-1.200, Rev. 2.

Internal Flood Events (ASME PRA Standard, Part 3)

The internal flood portion of the Seabrook PRA (IF PRA) has been updated several times, including the most recent 2011 revision of the 1991 IPEEE submittal (which was an update of the original SSPSA-1983). This latest update uses the most recent technical guidance for internal flood analysis.

A peer review was conducted in 2009 of the IF PRA, using two industry experts. The peer review resulted in a total of 32 F&Os, among these were 12 findings (F) associated with 16 SRs. The peer review identified five supporting requirements that were not met. These five SRs included: IFSO-A4 (F&O 5-13), IFQU-A7 (F&O 4-6), IFQU-B1 (F&O 4-7), IFQU-B2 (F&O 4-8) and IFQU-B3 (F&O 4-9).

A self-assessment of the internal flood PRA was conducted in 2010 following the revision to the IF PRA that addressed the 2009 Peer Review F&Os. The 2010 self-assessment addressed each of the 62 internal flood supporting requirements, based on a review of the 2009 Peer Review and an assessment of changes to the IF PRA. The 2010 SA identified no new observations or findings and determined that all IF SRs were met, with two exceptions (IFQU-A7 and IFQU-B1). However, these final open items were closed in the 2011 PRA Update. The 2014 PRA model updated the internal flooding

initiating event pipe break/flooding data and made some enhancements to flood HEP basic events to improve consistency within the model. These changes did not affect the status of any internal flooding SRs.

Conclusion - All Internal Flooding peer review findings are addressed and closed. The 2014 PRA fully meets all Part 3 (Internal Flood) CC II requirements (or higher) of the PRA Standard as endorsed by RG-1.200, Rev. 2.

Internal Fire Events (ASME PRA Standard, Part 4)

The internal fire portion of the Seabrook PRA (FPRA) has been updated several times, including the most recent 2004 revision of the 1991 IPEEE submittal (which was an update of the original SSPSA-1983). This recent update used then-current methods, but was performed before the most recent technical guidance in NUREG/CR-6850 (Reference 33).

The Seabrook FPRA has not been subject to formal peer review, but was independently reviewed by external experts in 2004. An independent gap assessment was performed in May 2014. The gap assessment concluded that, although the FPRA did not meet all of the Part 4 requirements of ASME/ANS RA-Sa-2009, the analysis provides meaningful qualitative and quantitative insights, and the resulting fire risk profile is consistent with similar design and vintage plants. In addition, Seabrook has initiated a number of modernization efforts to keep its FPRA current, including the partial draft update in 2008 and the MSO analysis of 2012. Those efforts generated significant technical analyses and while not yet implemented in the current model, represent an appreciable step toward compliance with the FPRA standard requirements.

As of this ILRT report, the Seabrook Station FPRA is in the process of being formally upgraded to meet the latest industry methods (guidance in NUREG/CR-6850) and CC II of the ASME PRA Standard. Insights gained thus far in the update process indicate that the current FPRA framework provides a realistic representation of fire risk. The FPRA upgrade may result in increased fire risk and/or may change the relative contributors to fire risk.

Conclusion - The Seabrook internal fire events PRA is a detailed quantitative model and is judged to provide a realistic/best estimate representation of plant fire risk. However, it has not been formally peer reviewed against the requirements of the PRA Standard and RG-1.200, Rev. 2. Accordingly, internal fire risk insights are qualitatively assessed (in Section 6.3 of the risk assessment) to further demonstrate that the proposed ILRT Type A test extension will have a minimal risk impact from fire events.

Seismic Events (ASME PRA Standard, Part 5)

The seismic events portion of the Seabrook PRA (SPRA) has been updated several times, including the most recent 2005 revision of the 1991 IPEEE submittal (which was an update of the original SSPSA-1983). This recent update used then-current methods to address issues related to equipment and operator fragility and the revised hazard spectrum but it did not include an update to the seismic hazard curve.

The Seabrook SPRA has not been subject to peer review, but was independently reviewed by external experts in 2005. An independent gap assessment was performed in May 2013. The gap assessment concluded that the Seabrook Station SPRA was judged to be in good accordance with the requirements of Part 5 of the ASME/ANS PRA standard and identified a number of improvements items related to modeling and quantification technical elements of the standard.

As of this report date, the current seismic model is judged to be sufficiently comprehensive to provide realistic quantitative and qualitative risk insights. A future action is to update the SPRA based on the latest GMRS and latest industry methods and insights from GI-199. The seismic hazard update may result in increased seismic risk and/or may change the relative contributors to seismic risk.

Conclusion - The SPRA was performed and revised using qualified analysts with then-current methods and it is judged to provide a reasonable representation of the seismic hazard risk from operation of Seabrook Station. However, the SPRA has not been formally peer reviewed against the requirements of the PRA Standard and RG-1.200, Rev. 2. Accordingly, seismic event risk insights are qualitatively assessed (in Section 6.3 of the risk assessment) to further demonstrate that the proposed ILRT Type A test extension will have a minimal risk impact from seismic events.

Other External Events (ASME PRA Standard, Parts 6 through 9)

The analysis of other hazards was performed for the original SSPSA-1983 and was updated for the 1991 IPEEE report. Most recently, the screening evaluation of other hazards was updated in March 2014. Specific evaluations of external events include:

Parts 6 and 9: Other External Hazards – Conservative Screening Analysis and PRA – As mentioned above, the screening evaluation of other external hazards was updated in March 2014. Other hazards includes all natural and man-made hazards that have the potential to impact Seabrook Station with the exception of specific hazards that are addressed in other sections of the PRA. The methodologies used for identification, characterization, qualitative screening, and quantitative assessment are consistent with the screening and assessment processes identified in the supporting requirements of Parts 6 and 7 of the ASME/ANS PRA Standard Addendum A, as endorsed by NRC Regulatory Guide 1.200 Rev 2. The scope of the other hazards review included assessment of the following hazards: industrial accident explosions, industrial accident toxic gas, meteorite impact, aircraft crash, transportation, turbine missiles, and heavy loads. These hazards are not significant contributors to plant risk and continue to screen based on conservative quantitative and qualitative screening criteria.

Part 7: High Winds PRA – The current PRA screens out high winds qualitatively based on the design consistency with the Standard Review Plan as addressed in the IPEEE Report (1992). In addition, high winds were addressed quantitatively in the original SSPSA (1983) with resultant CDF total ~1E-08 per year. Based on this assessment, the wind hazard is not a significant contributor to plant risk and continue to screen based on conservative quantitative and qualitative screening criteria.

Part 8: External Flood PRA - The current PRA screens out external flood hazard qualitatively based on the design consistency with the Standard Review Plan as addressed in the IPEEE Report (1992). The one exception to this screening is a single initiator EXFLSW, external flood event failing the ocean SW pumps to account for the risk of floods that could exceed the design basis. This hazard is explicitly included in the current internal events PRA model. This sequence was originally addressed quantitatively in the original SSPSA (1983), and currently has a CDF ~2E8 per year. A gap assessment of the External Flood PRA against the requirements in Part 8 of the PRA Standard was performed in 2012. This assessment summarized the current design basis flood information from the UFSAR as well as the potential beyond-design-basis floods from the Near-Term Task Force (NTTF) 2.3 walkdown report.

Conclusion - The Other Hazards screening analysis and PRA were performed and revised using qualified analysts with then-current methods and are judged to be a reasonable representation of the other hazards risk from operation of Seabrook Station. However, the other external events analysis has not been formally peer reviewed against the requirements of the PRA Standard and RG-1.200, Rev. 2. Accordingly, other external event risk insights are qualitatively assessed (in Section 6.3 of the risk assessment) to further demonstrate that the proposed ILRT Type A test extension has minimal risk impact from other external events.